Chapter 2
Program Description and Alternatives

Chapter 2 describes the program missions and the logic behind the structure of the program alternatives. Section 2.1 introduces the facility alternatives and options proposed to enhance the U.S. nuclear infrastructure and provide the capabilities needed to meet DOE's mission requirements. This section also presents the No Action Alternative and an alternative suggested by some of the public scoping comments. Sections 2.2 and 2.3, respectively, describe the operations necessary to implement DOE's program missions and the candidate sites and facilities where the operations would take place. Section 2.4 discusses the transportation activities associated with the program missions. Section 2.5 describes the alternatives that were considered reasonable for detailed evaluation. Section 2.6 explains why some other alternatives and facilities were considered and dismissed from evaluation in this NI PEIS. Section 2.7 summarizes the environmental impacts and implementation schedules associated with the alternatives that were evaluated and provides a comparative evaluation of these alternatives in terms of impacts and mission effectiveness. The chapter concludes with the description of the preferred alternative.

2.1 INTRODUCTION

As discussed in Chapter 1, the U.S. Department of Energy (DOE) is proposing to enhance its existing nuclear facility infrastructure to accommodate new and expanding missions in the areas of civilian nuclear energy research and development and isotope production. DOE currently does not have sufficient steady-state neutron sources to meet its projected irradiation needs for: (1) isotopes for medical and industrial uses, (2) plutonium-238 for potential use in advanced radioisotope (radioactive isotope) power systems and heating units for future U.S. National Aeronautics and Space Administration (NASA) space exploration missions, and (3) other irradiation services to meet the Nation's civilian nuclear energy research and development needs.

The programmatic alternatives focus on the use of irradiation facilities that are currently operating, could be brought on line, or could be constructed and operated to meet DOE's irradiation needs. Thus, this *Final Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility (Nuclear Infrastructure Programmatic Environmental Impact Statement* [NI PEIS]) evaluates the following alternatives:

- **Alternative 1**, resuming operation of the Fast Flux Test Facility (FFTF) at the Hanford Site (Hanford) in Washington State
- **Alternative 2**, using existing irradiation facilities (the Advanced Test Reactor [ATR] at Idaho National Engineering and Environmental Laboratory [INEEL], and the High Flux Isotope Reactor [HFIR] at Oak Ridge National Laboratory [ORNL], or a generic commercial light water reactor [CLWR])
- **Alternative 3**, constructing and operating one or two new accelerators at an existing DOE site
- **Alternative 4**, constructing and operating a new research reactor at an existing DOE site

This NI PEIS also evaluates a **No Action Alternative** in which the status quo would be maintained; that is, DOE's existing facilities would continue to meet their current mission requirements within their operating levels, but DOE would not enhance existing U.S. nuclear facility infrastructure or expand its current missions to accommodate new missions. This NI PEIS also includes an additional alternative that would permanently deactivate Hanford's FFTF without enhancing U.S. nuclear facility infrastructure to accommodate new or expanded missions. **Alternative 5**, permanently deactivate FFTF, although a component of all alternatives

except No Action and Alternative 1, is included as a stand-alone alternative in response to numerous public comments received during the scoping period.

This NI PEIS evaluates several options under each alternative. These options primarily involve DOE facilities that could be used for the fabrication, storage, and postirradiation processing of the targets necessary for the program missions. Among the facilities proposed are: (1) the Radiochemical Engineering Development Center (REDC) at ORNL, (2) the Fluorinel Dissolution Process Facility (FDPF) and/or Building CPP–651 (storage only) at INEEL, (3) the Fuels and Materials Examination Facility (FMEF) at Hanford, (4) the Radiochemical Processing Laboratory (RPL) and Building 306–E at Hanford, and (5) a new facility to be constructed and operated at an existing DOE site to support the one or two new accelerators or new research reactor alternatives. **Table 2–1** provides an overview of the alternatives and 26 specific options for this NI PEIS.

Sections 2.2 through 2.4 describe: (1) the operations needed to implement the program missions, (2) the candidate sites and facilities where the operations would take place, and (3) the transportation activities associated with the program missions. In describing the facilities, the sections refer to the specific alternatives for which the individual facilities are proposed. Detailed descriptions of alternatives are provided in Section 2.5. Alternatives considered and dismissed are discussed in Section 2.6. Section 2.7 summarizes the environmental impacts and implementation schedules and provides a comparison of mission effectiveness between alternatives, and Section 2.8 presents the preferred alternative.

2.2 DESCRIPTION OF OPERATIONS

2.2.1 Medical Isotopes Production

Production of medical and industrial isotopes involves: (1) fabricating specially designed targets at a target fabrication facility, (2) irradiating the targets in an irradiation facility to generate specific medical isotopes, and (3) processing the targets at a target fabrication facility to prepare the medical isotopes for shipment to customers.

2.2.1.1 Target Fabrication

Each medical isotope would be produced using a target that is enriched in the appropriate target material for neutron irradiation. With the exception of the radium-226 target and those that use recycled materials, all of the targets use nonradioactive materials. Appendix C lists the types and forms of the target material used to produce each medical isotope.

After irradiation of the original target, radioactive impurities may remain with the target material after the medical isotope product is removed. Because of these impurities, reuse of the material can in some cases create targets that are radioactive. As a result, fabrication of targets from recycled target materials would require special handling and shielding.

Before beginning fabrication of a target for production of a particular medical isotope, a significant quantity of the element that makes up the target would be required. For nonradioactive targets, this material typically would be acquired from ORNL, where enrichment processes are conducted to produce target material that is sufficiently pure to support the generation of medical isotopes. The target form may be a metal, metallic oxide, or other chemical compound suitable for high-temperature irradiation, depending on engineering considerations such as material heat transfer characteristics, melting points, and metallurgical properties. The nonradioactive target material would be transported by truck from ORNL to the target fabrication facility. One candidate radioactive target material, radium-226, would not be supplied by ORNL. However, radium-226 could be supplied by a variety of sources (no decision on a proposed supplier or suppliers has been made at

Table 2–1 NI PEIS Alternatives and Options

	Option Number	Irradiation Facility	Plutonium-238 Production Mission		Medical and Industrial Isotopes Production and Nuclear Research and Development Mission	
			Storage Facility	Target Fabrication and Processing Facility	Storage Facility	Target Fabrication and Processing Facility
No Action Alternative	1	–	–	–	–	–
	2	–	REDC	–	–	–
	3	–	CPP–651	–	–	–
	4	–	FMEF	–	–	–
Alternative 1: Restart FFTF	1	FFTF[a]	REDC	REDC	RPL/306–E	RPL/306–E
	2	FFTF[a]	FDPF/CPP–651	FDPF	RPL/306–E	RPL/306–E
	3	FFTF[a]	FMEF	FMEF	FMEF	FMEF
	4	FFTF[b]	REDC	REDC	RPL/306–E	RPL/306–E
	5	FFTF[b]	FDPF/CPP–651	FDPF	RPL/306–E	RPL/306–E
	6	FFTF[b]	FMEF	FMEF	FMEF	FMEF
Alternative 2: Use Only Existing Operational Facilities	1	ATR	REDC	REDC	–	–
	2	ATR	FDPF/CPP–651	FDPF	–	–
	3	ATR	FMEF	FMEF	–	–
	4	CLWR	REDC	REDC	–	–
	5	CLWR	FDPF/CPP–651	FDPF	–	–
	6	CLWR	FMEF	FMEF	–	–
	7	HFIR and ATR	REDC	REDC	–	–
	8	HFIR and ATR	FDPF/CPP–651	FDPF	–	–
	9	HFIR and ATR	FMEF	FMEF	–	–
Alternative 3: Construct New Accelerator(s)	1	New	REDC	REDC	New[c]	New[c]
	2	New	FDPF/CPP–651	FDPF	New[c]	New[c]
	3	New	FMEF	FMEF	New[c]	New[c]
Alternative 4: Construct New Research Reactor	1	New	REDC	REDC	New[c]	New[c]
	2	New	FDPF/CPP–651	FDPF	New[c]	New[c]
	3	New	FMEF	FMEF	New[c]	New[c]
Alternative 5: Permanently Deactivate FFTF (with No New Missions)	–	–	–	–	–	–

a. Hanford FFTF would operate with mixed oxide fuel for 21 years and highly enriched uranium fuel for 14 years.
b. Hanford FFTF would operate with mixed oxide fuel for 6 years and highly enriched fuel for 29 years.
c. The new facility would not be required if a DOE site is selected with available support capability and infrastructure.

Key: ATR, Advanced Test Reactor at INEEL; CLWR, commercial light water reactor; CPP–651, INEEL Building CPP–651 Storage Vault; FDPF, Fluorinel Dissolution Process Facility at INEEL; FFTF, Fast Flux Test Facility at Hanford; FMEF, Fuels and Materials Examination Facility at Hanford; HFIR, High Flux Isotope Reactor at ORNL; REDC, Radiochemical Engineering Development Center at ORNL; RPL/306–E, Radiochemical Processing Laboratory at Hanford, Building 306–E.

this time). Once materials for the targets arrive on the site, they would be stored at the target fabrication facility until needed for fabrication into medical isotope targets.

Solid targets would be fabricated in gloveboxes using a series of mechanical and thermal processes. For the solid targets based on a powder, it is unknown at this time whether the powder would be loose or would be pressed and sintered into pellets. If the latter method were preferred, separate equipment would be required to press and sinter each type of solid target material to reduce the risk of cross-contaminating other target materials.

If pellets were used, the first major step in their preparation would be powder conditioning and pressing, which includes weighing, blending, and pressing the powder and binder into slugs. The slugs would be granulated, blended with binder addition, and pressed into pellets. The pellets would be transferred to the sintering/debind station, weighed, and subjected to a series of thermal processes to debind and sinter the pellets. The sintered pellets would be subject to characterization to ensure that specifications were met.

Acceptable pellets would be transferred to the loading and welding station to be visually inspected before inclusion into a capsule or pin. For both powder or pellet target materials, capsules and pins would be cleaned before final closure. The capsules would be leak-tested and inspected before being cleared for use.

2.2.1.2 Target Irradiation

Production of medical or industrial isotopes is accomplished by irradiating target materials in the neutron flux of an irradiation facility such as a nuclear reactor. The desired isotopes are produced by neutron-induced reactions such as activation or transmutation.

Activation is the most common neutron-induced reaction and involves the capture of a neutron with the subsequent emission of a gamma ray. Because there is no change in the number of protons, the chemical identity of the target remains the same. For example, holmium-166 is produced by irradiating target material enriched in holmium-165 by activation.

Transmutation involves the capture of a neutron and the subsequent ejection of a proton or other particle that would change the chemical identity of the product. For example, phosphorus-32 is produced by irradiating a target material enriched in sulfur-32 by transmutation (proton ejection).

2.2.1.3 Postirradiation Target Processing

Processing of irradiated targets to recover medical- and industrial-grade isotopes can be broken down into distinct steps: (1) transport of irradiated targets to a chemical separation facility and receipt at that facility; (2) chemical processing of the targets (using hot cells, shielded gloveboxes and appropriate open-faced hoods); (3) waste handling; (4) analysis of the products; (5) recycling of some of the target materials; and (6) shipment of the isotope products to customers.

Each of the medical isotope products evaluated for production in this NI PEIS is unique. Some targets would produce an isotope of the same element and would not require separation. Some targets would produce the same element, but would require some processing to remove impurities. Other target materials would produce different elements and would require chemical separation of the target material, the desired isotope product, and unwanted impurities. Details on postirradiation processing of the targets for medical and industrial isotope production are provided in Appendix C.

2.2.2 Plutonium-238 Production

Production of plutonium-238 involves (1) storing neptunium-237, (2) fabricating neptunium-237 targets, (3) irradiating the targets in an irradiation facility, and (4) processing the targets to separate the plutonium-238 and prepare the product for shipment to Los Alamos National Laboratory (LANL) where it would be fabricated into heat sources for radioisotope power systems. As stated in Section 1.2.2, a plutonium-238 production rate of 2 to 5 kilograms (4.4 to 11 pounds) per year would be sufficient to meet the projected need based on NASA space exploration missions. Evaluations presented in this NI PEIS are based on a plutonium-238 production goal of 5 kilograms (11 pounds) per year to bound the environmental impacts of the proposed plutonium-238 production mission.

2.2.2.1 Target Fabrication

The facility designated to fabricate neptunium-237 target elements for plutonium-238 production would receive the neptunium-237 oxide from the Savannah River Site (SRS) and would dissolve it in an acid solution prior to removal of protactinium-233, a decay daughter of neptunium-237. Protactinium-233 reaches 90 percent of its equilibrium activity approximately 90 days after purification and contributes significantly to radiation doses in the target fabrication line. The best approach for the removal of the protactinium-233, and possibly the easiest to implement, is to pass the neptunium solution through a column containing silica gel adsorbent (Wham et al. 1998). After protactinium-233 removal, the purified neptunium solution can be transferred to a target-fabrication glovebox line and reconversion of the neptunium to the oxide form can be initiated. The desired form of the oxide (microspheres) is obtained by loading the neptunium on a cation-exchange resin of the selected particle size range, washing the loaded resin, and using heated air to oxidize the resin and form the neptunium dioxide microspheres.

Current target designs for the ATR and HFIR reactors consist of neptunium dioxide blended with aluminum powder, pressed into a target core, and clad with aluminum. This type of target has been used in nearly all of the DOE production and research reactors (except for fast neutron flux reactors, e.g., FFTF) to produce isotopes in general and plutonium-238 specifically.

Three different techniques can be employed to fabricate such targets:

1. The neptunium dioxide and aluminum powders are blended and pressed into pellets. The pellets are then loaded into aluminum target tubes, which are seal-welded and hydrostatically compressed.

2. The neptunium dioxide and aluminum powders are blended and pressed into compacts. The compacts are then roll-milled between aluminum cladding, after which the aluminum-clad neptunium dioxide is seal-welded.

3. The neptunium dioxide and aluminum powders are blended and pressed into billets and assembled into welded and evacuated aluminum containers. The billets and containers are then coextruded to produce target tubes.

All three techniques have advantages and disadvantages. The coextrusion technique has been used successfully by SRS in its plutonium-238 program and other special isotope programs. Demonstrations of the fabrication techniques would be required to determine which techniques are best for the proposed irradiation facilities.

The target blanket for the high-energy accelerator consists of neptunium dioxide blended with aluminum powder, pressed into the required configuration, and clad with aluminum.

The CLWR targets would have stainless steel or Zircaloy cladding because of the higher operating temperatures. Targets for the new research reactor would also have stainless steel (Incoloy-800) cladding for material compability. The postirradiation processing of these targets would be different from the postirradiation processing of the aluminum-clad targets.

The targets used for production of plutonium-238 in FFTF would be similar to target concepts developed in studies performed in 1992 and 1993. The proposed target assembly would consist of 19 large-diameter pins that combine alternating thin pellets or wafers of neptunium dioxide and yttrium hydride moderator within a ferritic alloy steel cladding. The steel cladding is required for compatibility with the reactor sodium.

The conversion efficiency of neptunium-237 to plutonium-238 during target irradiation at FFTF, ATR, HFIR, or CLWR would be approximately 10 to 15 percent. Approximately 50 kilograms (110 pounds) per year of neptunium-237 would be fabricated into targets to meet the plutonium-238 production goal of 5 kilograms (11 pounds) per year. Following processing of the irradiated targets, approximately 40 kilograms (88 pounds) per year of the unconverted neptunium-237 would be stored in liquid form and recycled for the fabrication of new targets. The remainder would be process waste. Based on the current preconceptual designs, the conversion efficiency of neptunium-237 to plutonium-238 during target irradiation at the new high-energy accelerator and new research reactor would be significantly lower, approximately 2.8 percent and 1.4 percent respectively. To meet the plutonium-238 production goal of 5 kilograms (11 pounds) per year, approximately 180 kilograms (396 pounds) would be fabricated into targets annually for irradiation in the high-energy accelerator, and 380 kilograms (836 pounds) would be fabricated into targets for irradiation in the new research reactor. The neptunium-237 target fabrication requirements for the new research reactor could be reduced by a factor of 4 to 8 by refining the current target and reactor preconceptual designs presented in Appendix E.

Neptunium must be treated like uranium-235 under DOE safeguards and security requirements, which are based on the mass of neptunium and the attractiveness level of the physical and chemical form of the neptunium. This would require special security clearances for persons with access to the neptunium, as well as ongoing security reviews and audits during the time of possession of the neptunium (McCallum 1999). Safe, secure trailer/SafeGuards Transports (SST/SGTs) would be required to transport any significant quantity of neptunium. The neptunium containers would be stored in specially designed storage vaults to provide a secure, safe storage for the materials. DOE guidelines concerning safeguards and security would be followed whenever materials were being stored or processed.

2.2.2.2 Target Irradiation

Irradiation of neptunium-237 targets in neutron flux produces plutonium-238 according to the following equations:

$$ {}_{93}Np^{237} + {}_{0}n^{1} \rightarrow {}_{93}Np^{238} $$

$$ {}_{93}Np^{238} \rightarrow {}_{-1}\beta^{0} + {}_{94}Pu^{238} $$

The neptunium-237 target nuclide absorbs a neutron to become neptunium-238 (first equation), which in turn decays with a half-life of 2.1 days and emits a beta particle (or electron) to form plutonium-238 (second equation).

Irradiation of the neptunium-237 targets generates fission products in the targets. The irradiated targets would be cooled for at least 120 days to allow time for the decay of short-lived fission products (e.g., iodine-131). Following the cool-down period, the irradiated targets would be loaded into a shielded cask for transport to

the chemical processing facility. They would then be ready for chemical processing to separate the plutonium-238 content and unconverted neptunium-237 from radioactive waste products.

2.2.2.3 Postirradiation Target Processing

The flowsheet for processing irradiated neptunium-237 targets, recovering the unconverted neptunium-237, refabricating target elements, separating the plutonium-238 product, and shipping the plutonium-238 for fabrication into heat sources for radioisotope power systems is shown in **Figure 2–1**. Processing the irradiated neptunium-237 targets would be conducted inside heavily shielded hot cells to protect workers from high radiation doses. Hot cells are specially designed shielded vaults or areas used for the remote handling and manipulation of some radioactive materials. Certain chemical processing steps would be required to recover the plutonium-238 as product and to recover the neptunium-237 for recycle. At ORNL and Hanford this process would be accomplished in two steps:

- For targets irradiated in ATR, HFIR, or the accelerator, caustic-nitrate solution would be used to dissolve the cladding, thereby separating the bulk of the aluminum and caustic-soluble fission products from the actinide products, including the neptunium and plutonium. For targets irradiated in FFTF, the new research reactor, or a CLWR, the first step in the target processing would be to chop the targets into small pieces for dissolution in acid (in the next step).

- Next, acid would be used to dissolve the actinide products and remaining fission products to prepare the feed for the first mainline separation process. The feed would be filtered prior to pH (acidity/alkalinity) adjustment to remove any solids that could complicate the solvent extraction process. For the FFTF, new research reactor, or CLWR targets, the undissolved cladding would be discarded as waste.

Dissolution of the irradiated targets at INEEL would be accomplished using a one-step target dissolution process in a nitric acid-fluoroboric acid solution instead of the two-step process for the ATR and HFIR targets that would be used at ORNL and Hanford. It would still be necessary to shear or chop the FFTF or CLWR targets at INEEL before the acid leach process.

Subsequent to target dissolution, a tributyl phosphate-based solvent extraction process would be used for three cycles of purification. The first cycle would decontaminate the neptunium and plutonium products from fission product wastes. The second cycle solvent extraction process would separate the neptunium from the plutonium, and the third-cycle process would remove trace plutonium from the neptunium product. The plutonium product would undergo further purification using anion exchange if the product did not meet specification.

Chemical conversion of the plutonium to an oxide would start with its precipitation from solution as an oxalate. The precipitate would be filtered and calcined (heated at high temperature) to an oxide product. The plutonium dioxide product would be further treated in an oxygen-exchange process to exchange its oxygen-17 and oxygen-18 components with oxygen-16, thereby reducing the neutron emission rate. The resulting oxide product would be packaged and shipped to LANL for fabrication into heat sources for radioisotope power systems.

The purified neptunium nitrate from the third-cycle solvent extraction process would be stored as a solution. A small quantity of neptunium oxide (6 to 8 kilograms [12 to 16 pounds]) would be removed from storage, dissolved, and purified to replace the neptunium-237 that was converted to plutonium-238. This material would be added to the neptunium solution recovered during postirradiation target processing, loaded onto a cation-exchange resin, and then calcined to produce oxide microspheres for re-use in target assemblies for

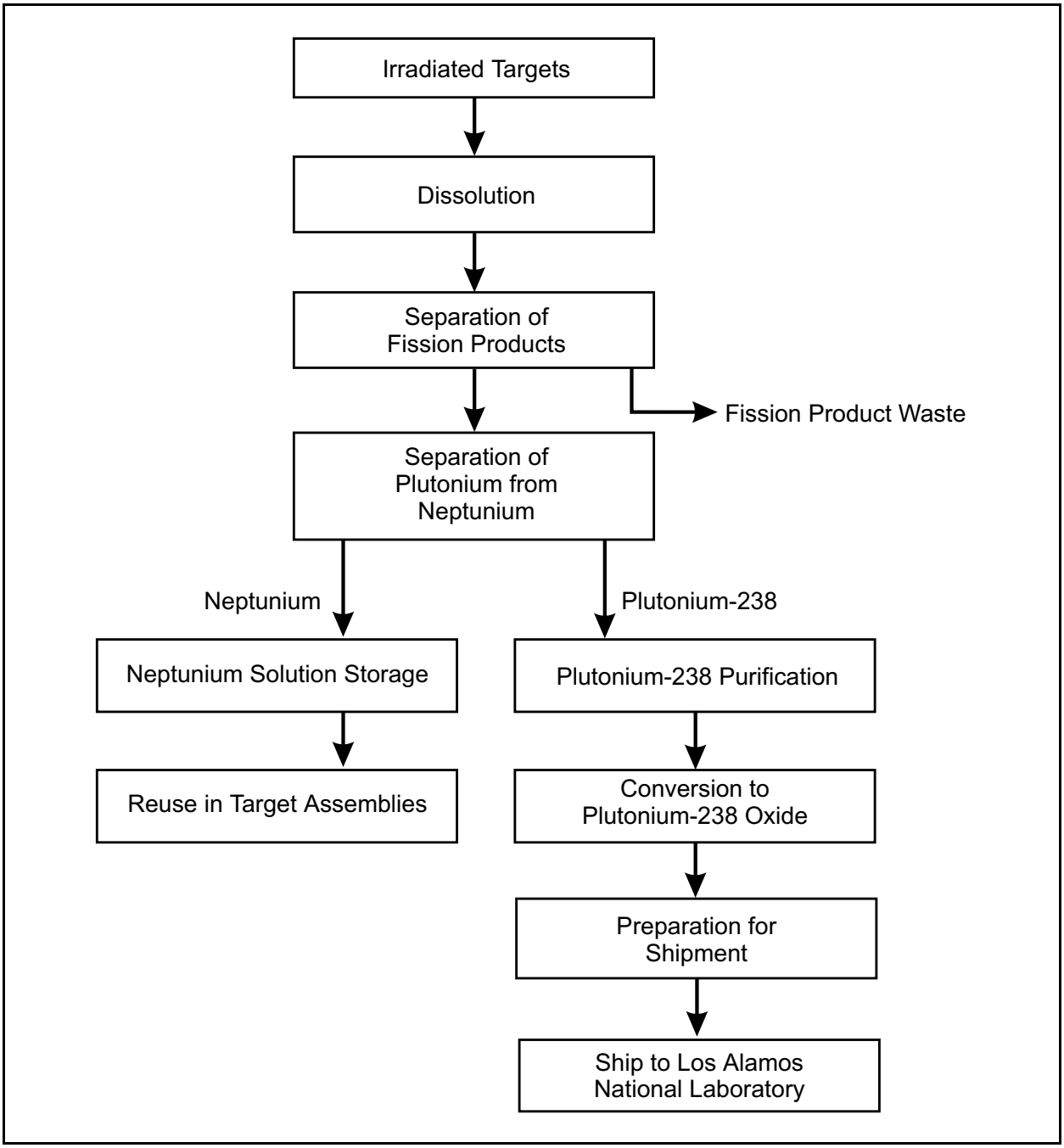


Figure 2–1 Chemical Processing Flowsheet for Irradiated Neptunium-237 Target Processing

irradiation. Waste-handling equipment would be used to minimize the activity in low-level radioactive liquid waste and to stabilize solid wastes into an acceptable waste form. The equipment would be included in the hot cells used for the chemical processing of irradiated targets.

2.2.3 Civilian Nuclear Energy Research and Development

As discussed in Section 1.2.3, civilian nuclear energy research and development initiatives requiring an enhanced DOE facility infrastructure fall into three basic categories: materials research, nuclear fuel research, and advanced reactor development.

2.3 DESCRIPTION OF FACILITIES

This section describes the facilities proposed by DOE for the production of medical and industrial isotopes, plutonium-238, and civilian nuclear energy research and development. Because the programmatic alternatives are structured around the use of the candidate irradiation facilities, they are discussed first in Section 2.3.1, followed by a discussion of the proposed target fabrication and processing facilities in Section 2.3.2. The proposed irradiation facility alternatives are (1) FFTF at Hanford, (2) ATR at INEEL, (3) HFIR at ORNL, (4) a generic CLWR, (5) one or two new accelerators at an existing DOE site, or (6) a new research reactor at an existing DOE site.

2.3.1 Target Irradiation Facilities

2.3.1.1 Fast Flux Test Facility

FFTF is a 400-megawatt thermal, liquid-cooled (sodium) nuclear test reactor (**Figure 2–2**) owned by DOE and located at Hanford in southeastern Washington State near Richland, Washington. Figure 3–12 presents a map of Hanford that depicts the location of FFTF. In May 1972, the U.S. Atomic Energy Commission published an environmental statement for FFTF (AEC 1972). That document provided information on the potential environmental impacts associated with the construction and operation of FFTF. In the late 1970s, the Safety Analysis Report prepared for FFTF was reviewed by the U.S. Nuclear Regulatory Commission (NRC) and the Advisory Committee for Reactor Safeguards. Comments from both organizations were addressed in the FFTF Final Safety Analysis Report. The construction of FFTF was completed in 1978.

Following extensive testing, FFTF was started up in April 1982. During its operation, FFTF successfully tested advanced nuclear fuels, materials, components, operating protocols, and reactor safety designs. It also produced a wide variety of medical isotopes and made tritium for the U.S. fusion research program.

Figure 2–2 Fast Flux Test Facility

FFTF was originally designed and operated as a science test bed for U.S. liquid metal fast reactor programs. These programs, which were canceled in 1993, were key elements both in closed fuel cycle and actinide waste disposition technology development. In December 1993, DOE decided not to operate FFTF because of a lack of

economically viable missions at that time. In accordance with the National Environmental Policy Act (NEPA), DOE published an environmental assessment and Finding of No Significant Impact for the shutdown and deactivation of FFTF in May 1995 (DOE 1995a). The environmental assessment contained an evaluation of the environmental impacts associated with the actions necessary to place FFTF in a radiologically and industrially safe shutdown condition suitable for long-term surveillance and maintenance before final decontamination and decommissioning.

The FFTF complex includes the reactor, as well as equipment and structures for heat removal, containment, reactor safety and shutdown systems core component handling and examination, fuel offloading and storage, utilities, and other essential services. There are 100 systems supporting various functions of FFTF during operation. The central structure of FFTF is the reactor containment building, an all-welded cylindrical steel structure 41 meters (135 feet) in diameter and 57 meters (187 feet) high. The array of buildings and equipment that surround the containment building and comprise the FFTF complex is shown in **Figure 2–3.** The reactor is below grade in a shielded cell in the center of the containment structure. Heat is removed from the reactor by circulating liquid sodium under low pressure through three separate closed primary piping loops, which include pumps, piping, and intermediate heat exchangers. These loops are located within inerted cells (cells filled with inert gases) within the containment structure. **Figure 2–4** is a cutaway of the containment building showing the location of the reactor, primary pumps, and intermediate heat exchangers. Three secondary sodium loops transport reactor heat from the intermediate heat exchangers to the air-cooled tubes of the dump heat exchangers. From there, the heat dissipates into the atmosphere through the forced draft dump heat exchanger. Commercial nuclear power reactors use reactor heat to create steam, which turns a turbine to produce electricity. FFTF, however, does not generate electricity.

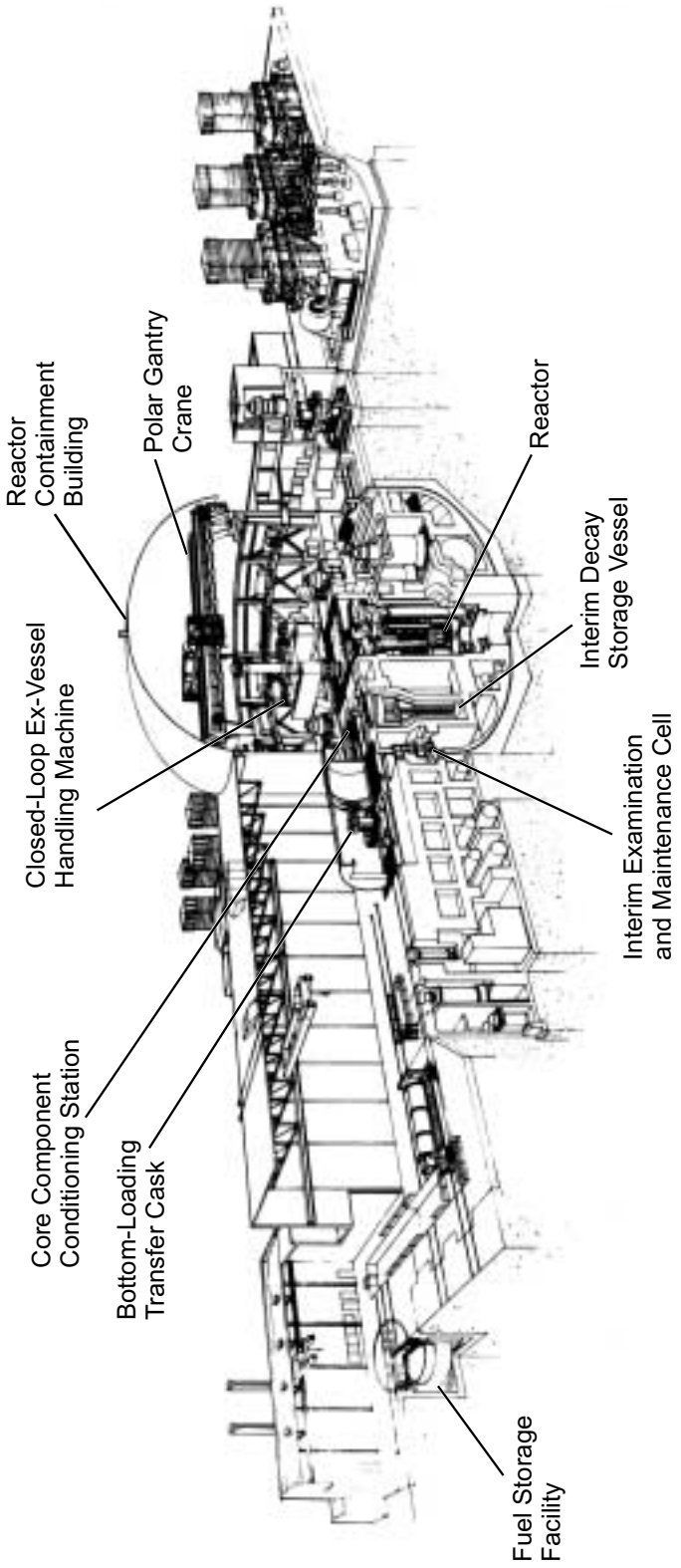
FFTF has demonstrated its capability to function as a nuclear science and irradiation services user facility. It has five distinct features: size, flux, test evaluation and irradiation capabilities, fuel type, and coolant type. In combination, these features provide a multipurpose facility suitable for medical and industrial isotopes production, plutonium-238 production, and civilian nuclear energy research and development purposes. Although FFTF was used primarily to evaluate reactor fuels and different fuel assembly materials during its 10 years of operation, the reactor facility has also supported large and varied test programs for industry, nuclear energy (domestic and international), medical isotope applications and research, space nuclear power, and fusion research programs. A more detailed description of FFTF and its capabilities is included in Appendix D.

2.3.1.1.1 Maintenance of FFTF in Standby

FFTF is currently defueled and is being maintained in a safe standby condition. FFTF would be maintained in the standby condition under the No Action Alternative. Seventy-seven of the 100 systems are operational; the other 23 are in a recoverable standby state. System integrity and configuration control are being maintained. The Main Heat Transport System is being operated at approximately 200 °C (400 °F) to keep the sodium coolant in the reactor liquefied and circulating. Essential systems, staffing, and support services are being maintained in a manner that would support potential restart.

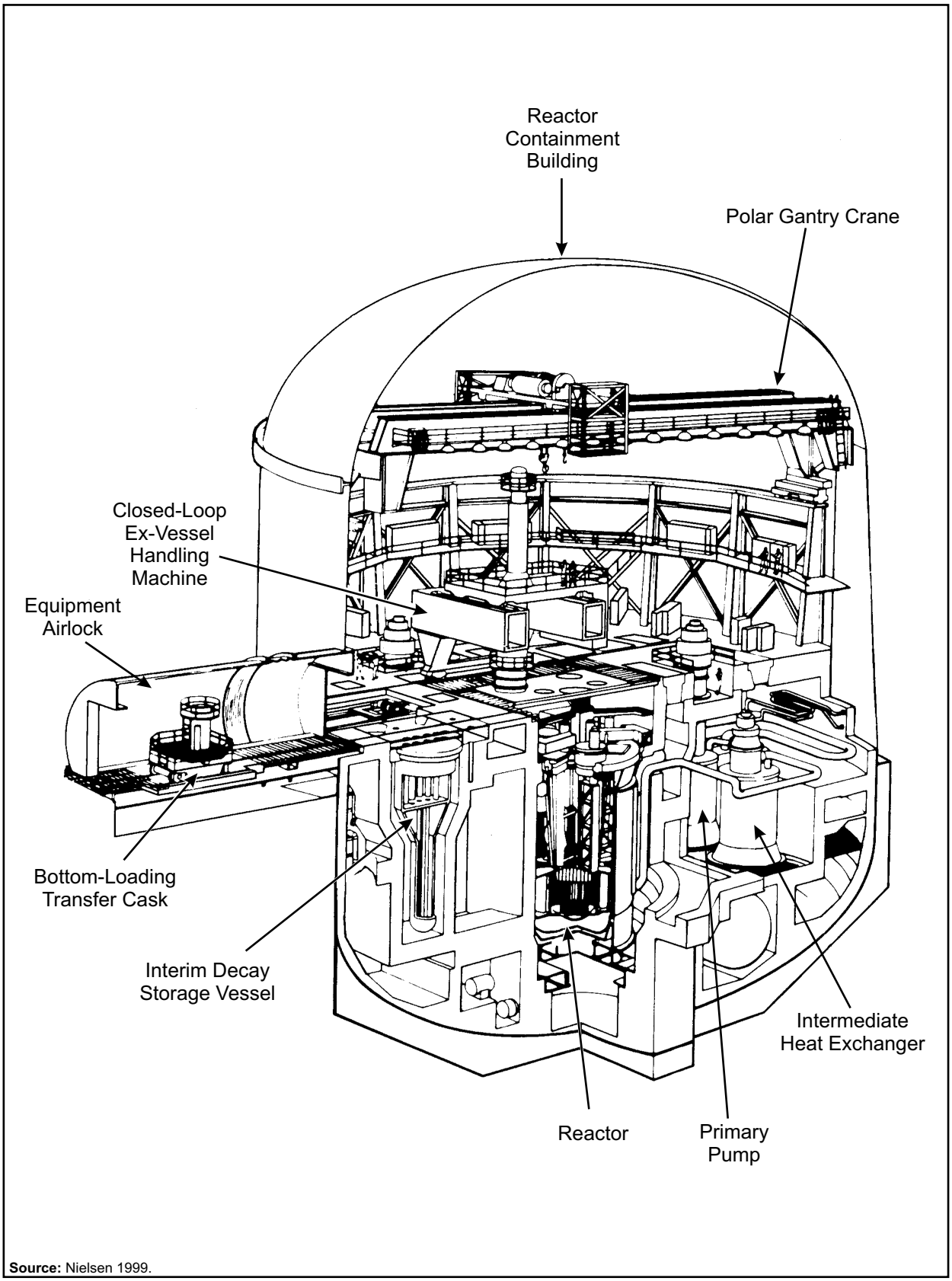
2.3.1.1.2 FFTF Restart and Operation

FFTF is proposed to be restarted and operated under Alternative 1, FFTF Restart. If a decision were made to restart FFTF, several equipment upgrades are planned to return systems to operation, improve reliability, conform to current standards, improve efficiency, and minimize waste. Most of the required modifications would consist of either mechanical equipment upgrades or replacement of outdated control and computer



Source: Nielsen 1999.

Figure 2–3 FFTF Complex



Source: Nielsen 1999.

Figure 2–4 Cutaway of the Reactor Containment Building

systems and would have minimal environmental consequences. The following is a brief list of the planned modifications if FFTF would be restarted (Battelle 1999):

- Upgrade of plant protection system (scram breakers, power supplies, and signal conditioners)
- Replacement of zero-time-outage motor generator sets with solid-state electronic units
- Upgrades of plant data systems computers
- Upgrade of conductivity metering system on three cooling towers and replacement of electronic sensors and controls
- Installation of two new electrical distribution transformers to replace the polychlorinated biphenyl-filled units that were removed during standby operations
- Establishment of a program to assess and replace elastomer seals during the startup period to take advantage of advancements in seal technology
- Upgrades of the plant simulator (A program to upgrade the existing simulator to reach commercial simulator standards was in progress, but was discontinued when FFTF was placed in standby.)

2.3.1.1.3 FFTF Fuel Use Option

This NI PEIS postulates that FFTF would operate at a nominal power level of 100 megawatts, one quarter of the reactor design power level, to meet the irradiation requirements of the proposed missions. Periodic increases in power level between 100 and 400 megawatts may be required to support civilian nuclear energy research and development activities. Operating FFTF at a nominal 100-megawatt power level would extend the reactor life and significantly reduce the generation rate of spent fuel. FFTF is currently designed to operate using mixed oxide fuel (i.e., plutonium-uranium); however, it can also be operated using highly enriched uranium fuel. FFTF has an onsite supply of mixed oxide fuel for approximately 6 years of operation at the 100-megawatt level proposed for the mission. When this onsite fuel is depleted, FFTF may continue to use mixed oxide fuel or may switch to a reactor core of highly enriched uranium fuel. DOE believes that an additional 15-year supply of mixed oxide fuel would be available from Germany under favorable economic terms, (i.e., no charge for the fuel). The fuel would be reconfigured into assemblies suitable for irradiation at FFTF before shipment to the United States. That is why this NI PEIS evaluates the operation of FFTF for two reactor core configurations for the 35-year evaluation period of operation common to all alternatives: (1) operation with a mixed oxide core for approximately 21 years followed by 14 years of operation with a highly enriched uranium core, and (2) operation with a mixed oxide core for approximately 6 years followed by 29 years of operation with a highly enriched uranium core.

In this NI PEIS, DOE has not evaluated the possibility of using low-enriched uranium fuel for operation of the FFTF because it makes programmatic and economic sense to use available mixed oxide fuel supplies before using uranium. U.S. nonproliferation policy (U.S. House of Representatives 1992 [Schumer Amendment]), strongly discourages the use of highly enriched uranium fuel in civilian research and test reactors. The Reduced Enrichment for Research and Test Reactors Program implements this policy by developing technical means to reduce and eventually eliminate the use of highly enriched uranium in research and test reactors throughout the world and in the United States without decreasing their safety or significantly affecting their performance and operating costs.

To be in compliance with these policy directives, the most appropriate fuel supply for FFTF in the out-years (beyond current Hanford mixed oxide and possible German SNR–300 mixed oxide supplies) must be determined by a technical study with the preferred fuel source being low-enriched uranium. Highly enriched uranium fuel should only be considered if low-enriched uranium is not technically feasible, or if there are significant impacts on safety, performance, or cost associated with using fuels other than highly enriched uranium.

In the event that a decision is made to restart the reactor, and to support these policy directives, DOE's Office of Nonproliferation and National Security would undertake a study to consider the technical feasibility of using low-enriched uranium fuel (under the Reduced Enrichment for Research and Test Reactors Program) for FFTF. If low-enriched uranium fuel is found infeasible, DOE would subsequently procure highly enriched uranium fuel in a manner consistent with U.S. nonproliferation policy. This study would be conducted, decisions would be implemented, and fuel would be made available during the time period between a Record of Decision indicating an FFTF restart and prior to the end of available Hanford mixed oxide and possible SNR–300 mixed oxide fuel supplies.

For the purposes of presenting a bounding analysis in this NI PEIS, DOE has analyzed the impacts of using highly enriched uranium fuel in FFTF after the available mixed oxide fuel supplies have been expended. These impacts would bound those of using a low-enriched uranium fuel form.

2.3.1.1.4 FFTF Irradiation Operations

There are eight locations available in the reactor core that are termed Open Test Assembly positions. These positions are located under spool pieces in the reactor head and allow the installation of 38-foot-long assemblies that extend from the reactor head down to the reactor core. These eight locations are unique from the rest of the reactor in that they allow direct contact instrumentation for remote monitoring during reactor operation. Within the 82 active core locations, there are up to 20 or more additional locations that could contain a standard length (3.6-meter or 12-foot) test assembly. These locations also have specific online outlet temperature and flow measurements from installed plant instrumentation. In addition to the test locations within the active fueled region of the core, there are 108 locations available in the surrounding reflector region where other tests could be inserted. These three basic testing configurations enable irradiation of large and/or very diverse quantities. The target designs vary according to the test requirements and the location of the test within the reactor.

To fulfill the mission, the FFTF core would be modified to include an array of target assemblies and rapid radioisotope retrieval systems capable of producing a number of long- and short-lived isotopes for medical and industrial applications and plutonium-238 for space power applications. In addition, reactor space would be provided for research and development test articles.

Fifteen plutonium-238 production targets would be included in the reflector region with an annual production rate of 5 kilograms. The residence time for these targets would be three 100-day cycles; five assemblies would be harvested at the end of each cycle.

Long-Term Irradiation Vehicles would be used to irradiate targets to produce long-lived isotopes. The Long-Term Irradiation Vehicles would be installed in the reactor during normal refueling operations and would be handled using standard FFTF component handling equipment. The Long-Term Irradiation Vehicle would consist of a bundle of target pins installed inside a nozzle, duct, and handling socket assembly similar in appearance to an FFTF 3.6-meter-long (12-foot-long) fuel assembly. Depending on the isotopes to be produced, the pin bundle could contain moderator pins and neutron shield pins. A design that would allow re-use of the long-term irradiation assembly nozzle, duct, and handling socket hardware would be considered

during the design process to reduce both costs and waste generation. It is assumed that 12 Long-Term Irradiation Vehicle assemblies for the production of long-lived medical and industrial isotopes would be installed. A detailed description of the Long-Term Irradiation Vehicles and their proposed use is included in Appendix D.

Rapid radioisotope retrieval systems would be installed in selected Open Test Assembly positions for the production of short-lived isotopes. These systems would extend from above the spool pieces in the reactor head down into and slightly below the active core region and would allow target materials to be inserted and withdrawn from the reactor core region while the reactor is operating. Systems for routinely inserting and removing irradiation targets, nuclear instrumentation, and research hardware have been in use for years at various research reactors throughout the world. Most of these systems use either a pneumatic rabbit-type system or a mechanical cable-type system for insertion and retrieval. There would be a maximum of eight systems in the core. One of the systems would be configured as a gas target to produce iodine-125 from xenon-124. The other seven systems would be used to produce solid short-lived medical isotopes. A detailed description of the rapid radioisotope retrieval systems and their proposed use is included in Appendix D.

FFTF would operate at a nominal power level of 100 megawatts. However, the accident analyses provided in this NI PEIS are based on the FFTF design power level of 400 megawatts and provide conservative estimates of operation at 400 megawatts-thermal and lower power levels.

Testing programs would be conducted for new materials and target designs to be irradiated in the reactor. A discussion of the types of testing that would be associated with the medical isotope and plutonium-238 production missions is included in Appendix D.

2.3.1.1.5 FFTF Deactivation

FFTF would be permanently deactivated in Alternative 2 (Use Only Existing Operational Facilities), Alternative 3 (Construct New Accelerator[s]), Alternative 4 (Construct New Research Reactor), and Alternative 5 (Permanently Deactivate FFTF [with No New Missions]). This would require placement of FFTF in a radiologically and industrially safe shutdown condition that is suitable for a long-term surveillance and maintenance phase prior to final decontamination and decommissioning. An *Environmental Assessment, Shutdown of the Fast Flux Test Facility, Hanford Site, Richland, Washington*, issued by DOE in 1995, addressed the environmental impacts associated with permanently deactivating FFTF (DOE 1995a).

If a decision were made to proceed with permanent deactivation of FFTF, the molten sodium (radioactive) would be removed from the reactor systems and transferred to an existing sodium storage facility that was specially constructed for this purpose. The sodium would be drained by pressure transfer to the maximum practical extent into tanks located in the sodium storage facility. Residual sodium would be accommodated to a stabilized condition so that long-term monitoring and surveillance could be conducted in a safe and environmentally sound manner. The current concept for accommodating residuals would be to maintain an inert gas atmosphere that prevents any chemical reactions during long-term surveillance and maintenance.

2.3.1.2 Advanced Test Reactor

ATR is a light-water-cooled and moderated reactor with a design thermal power of 250 megawatts that is owned by DOE and is in the Test Reactor Area in the southwest portion of INEEL. Figure 3–6 presents a map of INEEL that depicts ATR's location.

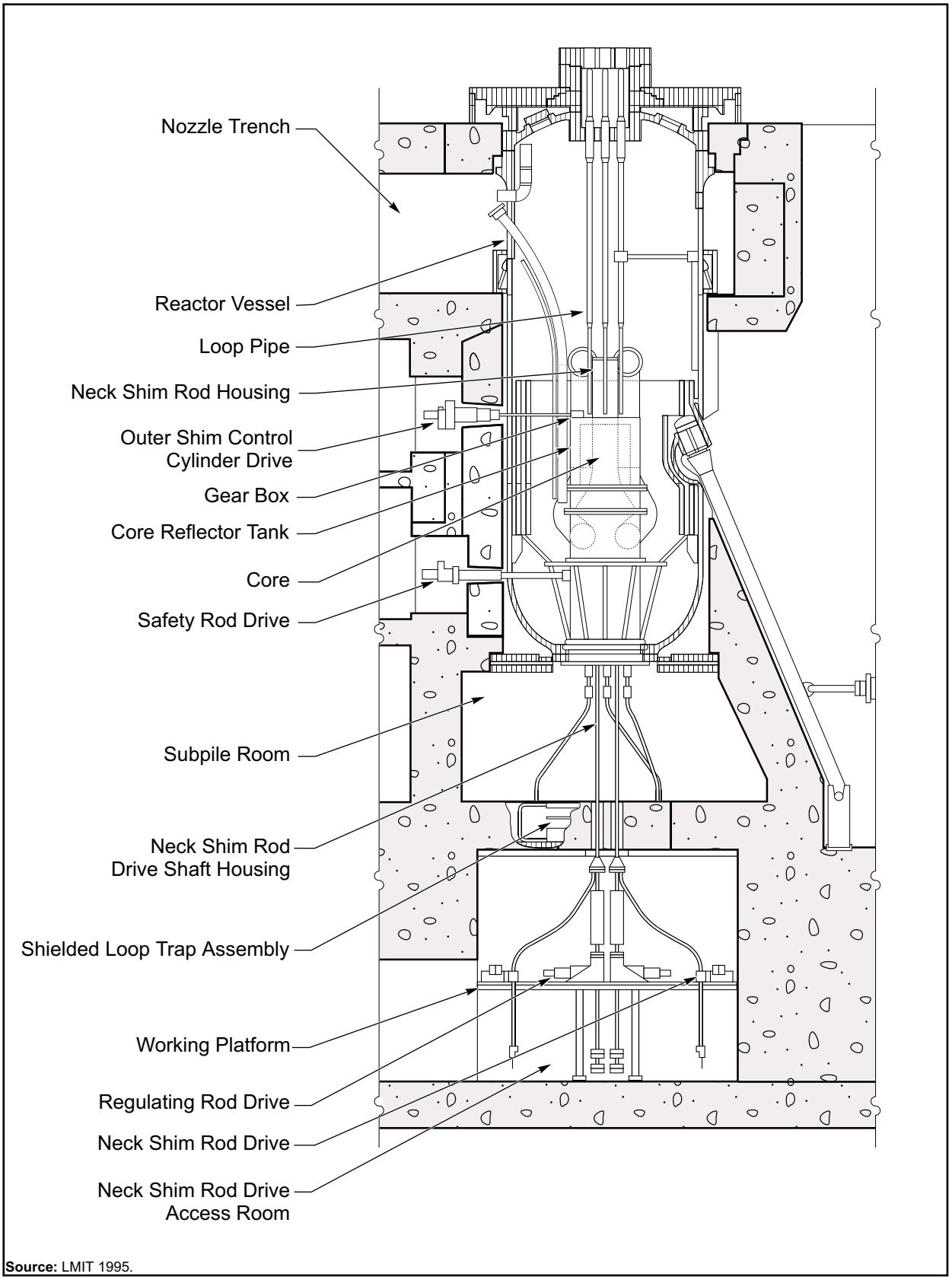
ATR would continue to operate and meet its current mission requirements, including naval reactor research and development, medical and industrial isotope production, and civilian nuclear energy research and

development activities, at its current operating levels under the No Action Alternative, Alternative 1 (FFTF Restart), Alternative 3 (Construct New Accelerator[s]), Alternative 4 (Construct New Research Reactor), Alternative 5 (Permanently Deactivate FFTF [with No New Missions]), and Alternative 2 (Use Only Existing Operational Facilities) when it is not providing irradiation services in support of the plutonium-238 production mission. When ATR is supporting the plutonium-238 production mission, it would fully support its primary mission—naval reactor research and development; however, it would support the medical and industrial isotope production and civilian nuclear energy research and development activities to the extent possible within its current reactor operating levels. Consideration must be given to the need to maintain appropriate levels of neutron flux to support ATR's primary mission. Neutron flux levels can be impacted by the placement of targets, such as neptunium-237 targets for the production of plutonium-238, in the reactor core. The production planning assumption for ATR is from 3 kilograms (6.6 pounds) of plutonium-238 per year (if used in conjunction with HFIR) to 5 kilograms (11 pounds) of plutonium-238 per year (if ATR were used alone). Thus, ATR alone could meet the program goal of up to 5 kilograms (11 pounds) per year and could be used in combination with any one of the three processing facilities for the plutonium-238 production mission.

Special features of ATR include high neutron flux levels (ranging from 1×10^{15} neutrons per square centimeter per second in the flux traps to 1×10^{13} neutrons per square centimeter per second in the outer reflector positions) and the ability to vary power to fit different experiment needs in different test positions. The primary user of ATR is the U.S. Naval Nuclear Propulsion Program. A variety of other users include other foreign and domestic government programs, a commercial isotope production company, industrial customers, and research and development interests. This facility description is based on information provided in the *Advanced Test Reactor Upgraded Final Safety Analysis Report* (LMIT 1997) and *Capabilities of the Test Reactor Area Featuring the Advanced Test Reactor* (LMIT 1995). A number of support facilities are important to the operation of ATR (LMIT 1997). Among these are the Advanced Test Reactor Critical Facility, which is used to baseline experiment impacts to the ATR flux profile, and the Nuclear Materials Inspection and Storage Facility, which is used to receive, store, and inspect reactor fuel prior to its placement in ATR (INEEL 1999, 2000; LMIT 1995).

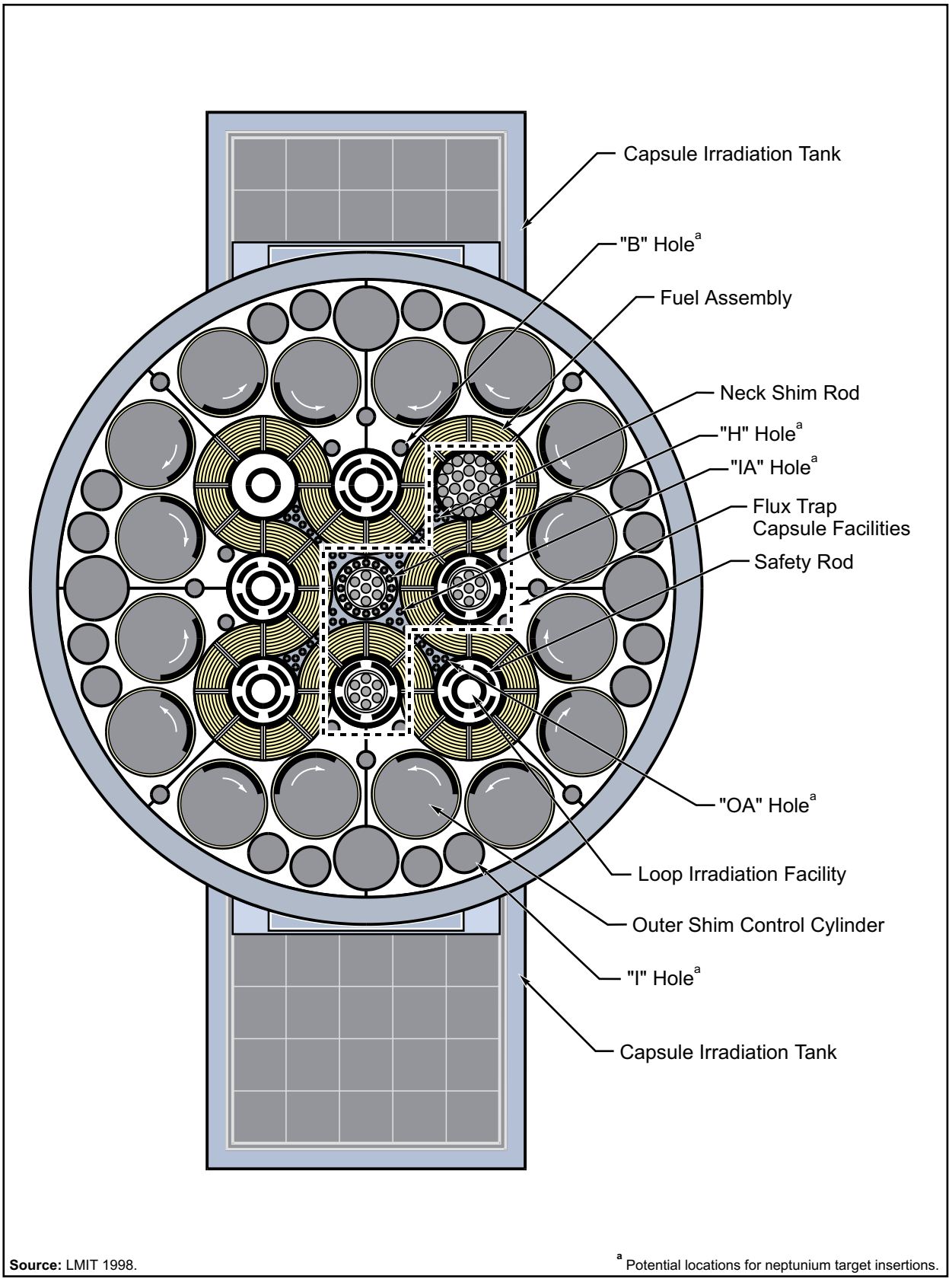
The reactor, its primary coolant system, control room, and much of its auxiliary and experimental support equipment are in Test Reactor Area Building 670. ATR began operation in 1967 and is expected to continue operating for several decades. The reactor vessel is constructed entirely of stainless steel, and the core internals are replaced every 7 to 9 years. The most recent changeout was completed in 1994 (LMIT 1995). Buildings and structures in other parts of the Test Reactor Area provide additional support functions.

ATR is currently operating at approximately 140 megawatts or less. ATR operates with highly enriched uranium fuel. Typical operating cycles are 42 days or 49 days at power followed by a 7-day outage for refueling and changeout of experiments and isotope production targets. The core is 1.2 meters (4 feet) high and is surrounded by a 1.3-meter-diameter (4.25-foot-diameter) beryllium reflector. Beryllium is an excellent neutron reflector and is used to enhance the neutron flux essential to a test reactor. The location of the core in the ATR vessel is shown in **Figure 2–5**. ATR has nine flux traps in its core and achieves a close integration of flux traps and fuel by means of a serpentine fuel arrangement (**Figure 2–6**). When viewed from above, the ATR fuel region resembles a four-leaf clover. The four flux traps positioned within the four lobes of the reactor core are almost entirely surrounded by fuel, as is the center position. Four other flux trap positions between the lobes of the core have fuel on three sides. ATR's unique control device design permits large power shifts among the nine flux traps. Testing can be performed in test loops installed in some flux traps with individual flow and temperature control or in reflector irradiation positions with primary fluid as coolant. The curved fuel arrangement brings the fuel closer on all sides of the test loops than is possible in a rectangular grid.



Source: LMIT 1995.

Figure 2–5 Vertical Cross Section of the ATR Vessel



Source: LMIT 1998.

[a] Potential locations for neptunium target insertions.

Figure 2–6 Plan View (Cross Section) of ATR

Five out of the nine flux traps are configured with pressurized-water loops that allow for individual temperature, pressure, flow, and chemistry controls. The five test loops are used by the Naval Reactors program. Of the remaining four flux traps, one is dedicated to the Naval Reactors program, one is used for isotope production, one is used for low-specific-activity cobalt production, and the fourth has recently had the Irradiation Test Vehicle installed. The Irradiation Test Vehicle can be described as three small pressurized-gas test loops. The use of one of these three test loops was recently purchased by a British corporation; negotiations for use of the other two are currently under way.

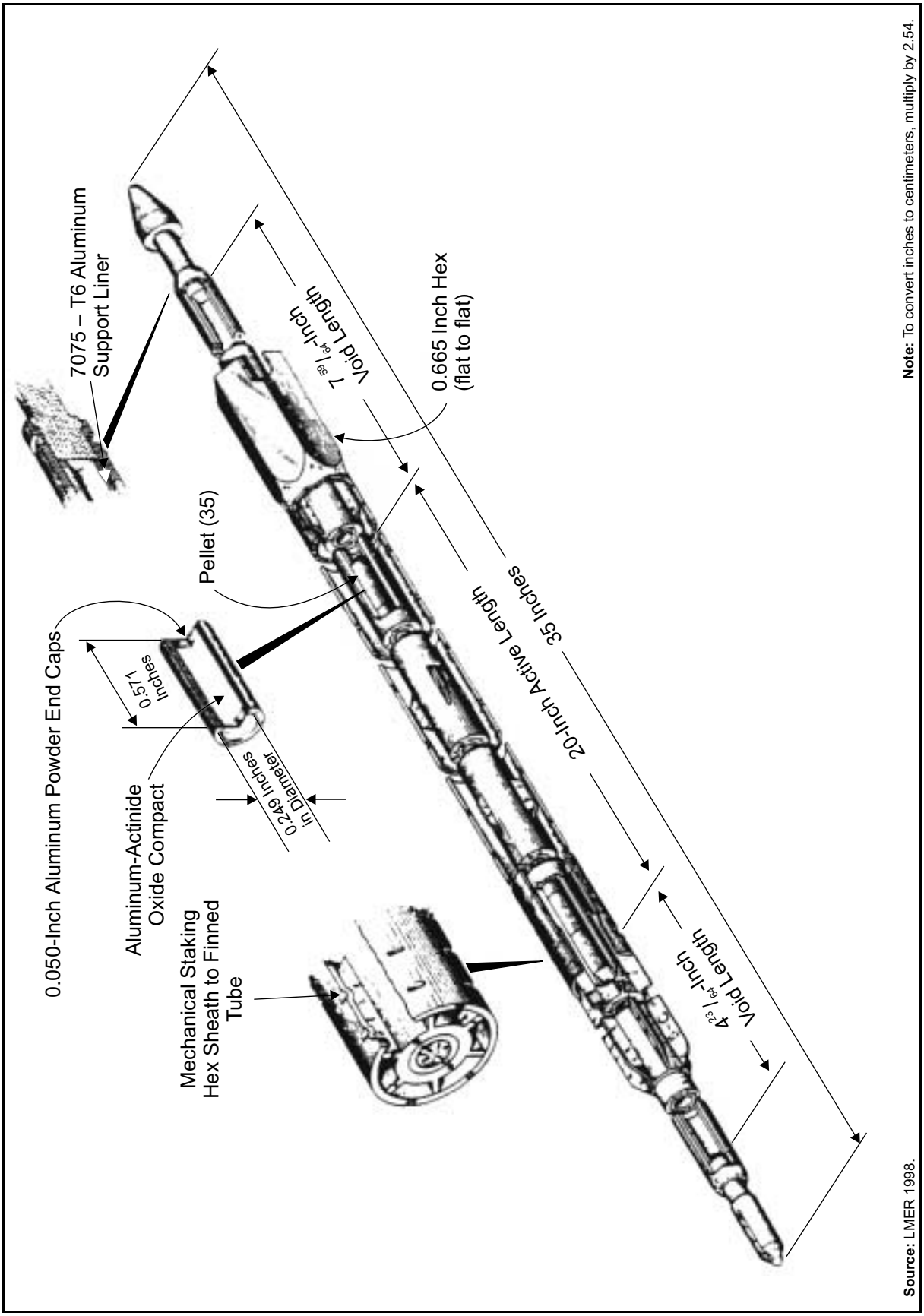
In addition to the primary flux trap irradiation positions, there are some 70 irradiation positions in the beryllium reflector (and aluminum support structure) that are available for experiment irradiation and isotope production. These position diameters range from 1.6 centimeters (0.625 inch) to 12.7 centimeters (5 inches) with thermal neutron flux levels ranging from 1×10^{15} neutrons per square centimeter per second to 1×10^{13} neutrons per square centimeter per second. Approximately 25 percent of the high-flux test positions (A holes, B holes, and H holes) are currently used for iridium-192 production. The majority of the remaining high-flux test positions are used for cobalt-60 production. Occasionally, additional isotopes (e.g., strontium-89, nickel-63) are generated in small quantities. A private company leases the space for the production of these isotopes. A small number of positions are used by other companies or government programs for other materials irradiation projects. For the production of plutonium-238, neptunium-237 targets would be placed in the beryllium reflector positions. The proposed target design consists of neptunium dioxide blended with aluminum powder, pressed into a target core, and clad with aluminum. The basic ATR target should be similar in appearance to, but longer than, the typical transuranic isotope production target shown in **Figure 2–7**. The ATR target length would be sized for the 1.2-meter (4-foot) active core length of ATR. Beryllium reflector position sizes range from 1.6 centimeters (0.625 inch) in diameter to 12.7 centimeters (5 inches) in diameter.

ATR is equipped with numerous safety features, including extensive plant protective systems, standby power sources, experiment interlocks, computerized surveillance, confinement systems, safety rods, and an emergency firewater injection system. ATR's six safety rods provide fast shutdown of the reactor if potentially damaging conditions develop. A sudden rise in power or coolant temperature, a sudden drop in coolant flow or pressure, or the overheating of a test sample are examples of approximately 360 conditions that would automatically drop the safety rods into the core. The firewater injection system provides emergency core cooling and flooding of the reactor vessel in the event of a loss of primary coolant. ATR is connected by a water canal to the ATR Critical Facility. The ATR Critical Facility is a low-power, full-size nuclear duplicate of ATR that is used to provide data as needed for experiment loadings prior to irradiation of the actual experiments in ATR.

INEEL has privatized the production of medical and industrial isotopes through contracting with a commercial entity. International Isotopes Idaho, Inc. (I^4), was selected in October 1996 as the commercial business for conducting these business operations. I^4 specializes in producing isotopes targets for irradiation in ATR and processing and distributing commercial-grade isotopes to their customers. Prior to commercialization, INEEL's isotope production operations were limited in types and quantities. Since the start of commercial activities, I^4 has expanded its commercial production to become a major world supplier of several important isotopes. I^4 has doubled the use of ATR irradiation positions for this purpose.

The major isotopes currently produced by INEEL and I^4 are iridium-192, 70 percent of the total U.S. demand; cobalt-60, 95 percent of the U.S. medical market; strontium-89, only U.S. supplier; and nickel-63, only U.S. supplier and producer of 50 percent of the world market.

Incremental investments have been identified for ATR that would make it a more versatile and capable reactor for isotope production. I^4 and another commercial company are in the discussion phase of investing in ATR to install an isotope rabbit (shuttle) system for the production of short-lived radioisotopes. Many of these



Source: LMER 1998.

Note: To convert inches to centimeters, multiply by 2.54.

Figure 2–7 Typical Transuranic Isotope Production Target

short-lived radioisotopes are expected to be in growing demand for various cancer therapies. I^4 has also committed to keep part of the rabbit system available for other users.

2.3.1.3 High Flux Isotope Reactor

HFIR is a beryllium-reflected, light-water-moderated and -cooled reactor operating at a thermal power level of 85 megawatts. HFIR is owned by DOE and is in the 7900 Area of the ORNL site in the southern portion of the Oak Ridge Reservation (ORR). Figure 3–1 presents a map of ORR that depicts HFIR's location.

HFIR would continue to be operated to meet its primary mission of neutron science-based research for the DOE Office of Science. In addition, medical and industrial isotope production and civilian nuclear energy research and development activities would be performed on a not-to-interfere basis at the current operating level in the No Action Alternative, Alternative 1 (FFTF Restart), Alternative 3 (Construct New Accelerator[s]), Alternative 4 (Construct New Research Reactor), Alternative 5 (Permanently Deactivate FFTF [with No New Missions]), and Alternative 2 (Use Only Existing Operational Facilities). When HFIR is supporting the plutonium-238 production mission, it would fully support its primary mission, but would support the medical and industrial isotope production and civilian nuclear energy research and development activities to the extent possible within the current reactor operating levels. Consideration must be given to the need to maintain appropriate levels of neutron flux to support HFIR's primary mission. Neutron flux levels can be impacted by the placement of targets (such as neptunium-237 targets for the production of plutonium-238) in the reactor core. Under the planning assumptions for plutonium-238 production, HFIR could only produce from 1 to 2 kilograms (2.2 to 4.4 pounds) per year without impacting ongoing missions. As the program goal is to achieve a production rate of 5 kilograms (11 pounds) per year, production from HFIR would need to be augmented by the use of ATR to meet this goal. HFIR and ATR together could meet the program goal of up to 5 kilograms (11 pounds) per year and could be used in combination with any one of the three processing facilities for the plutonium-238 production mission.

HFIR was originally designed as both an isotope production and a research reactor with a thermal flux of 3 to 5×10^{15} neutrons per square centimeter per second and a full power level of 100 megawatts-thermal (3.4×10^8 British thermal units per hour). It is currently operating at a maximum authorized power level of 85 megawatts-thermal (2.9×10^8 British thermal units per hour) to extend the useful life of the reactor. Many experiment-irradiation facilities were provided for in the original design and several others have been added. The primary mission of HFIR is neutron science research. Isotope production is done on a not-to-interfere basis.

HFIR transfers its primary coolant heat load to secondary coolant through heat exchangers for dissipation to the atmosphere by an induced-draft cooling tower. The reactor uses highly enriched uranium and aluminum-clad plate fuel. The reactor vessel itself is immersed in a pool in a poured-concrete reactor building that also houses the primary coolant pumps and heat exchangers, a spent fuel pool, and experiment areas. The control and water wing of the reactor building contains the reactor control room; relay and amplifier areas; heating and ventilating equipment; pool and fire alarm equipment; instrumentation systems; and office and support rooms. A separate electrical building adjacent to the reactor building contains switchgear, diesel generators, and associated transformers that connect the facility to offsite power. The reactor building is essentially airtight and provides dynamic confinement. A special hot exhaust system exhausts air from potentially contaminated areas of the building through filters (two HEPA filters and two charcoal filters) before being released to the atmosphere through a 76-meter (250-foot) stack. The stack serves as the exhaust point for both HFIR and REDC at ORNL.

After the reactor completed 17.2 full-power years of its 20 full-power year design life in November 1986, several measures were taken to extend the useful life of the reactor, including reducing the

100 megawatts-thermal (3.4×10^8 British thermal units per hour) rated power level to 85 megawatts-thermal (2.9×10^8 British thermal units per hour); adjusting the primary coolant temperature and pressure; conducting periodic hydrostatic tests; establishing an irradiation embrittlement surveillance program; and installing an emergency depressurization system. Subsequent life extension programs can enable HFIR to provide support during the total 35-year evaluation period for operations.

A plan view of the reactor (**Figure 2–8**) provides a cross section of the reactor vessel depicting experiment irradiation capabilities. Available experiment irradiation facilities include (1) the hydraulic tube facility in the very high flux region of the flux trap that allows for insertion and removal of irradiation samples while the reactor is operating; (2) 30 target positions in the flux trap that normally contain transuranium production rods but can be used for the irradiation of other experiments (two instrumented target positions were provided by a recent modification); (3) six peripheral target positions at the outer edge of the flux trap; (4) numerous vertical irradiation facilities of various sizes located throughout the beryllium reflector; (5) two pneumatic tube facilities in the beryllium reflector that allow insertion and removal of irradiation samples while the reactor is operating for activation analysis; (6) four horizontal beam tubes that originate in the beryllium reflector; and (7) four slant access facilities, called "engineering facilities," located adjacent to the outer edge of the beryllium reflector. In addition, spent fuel assemblies are used for gamma irradiation in the gamma irradiation facility in the reactor pool.

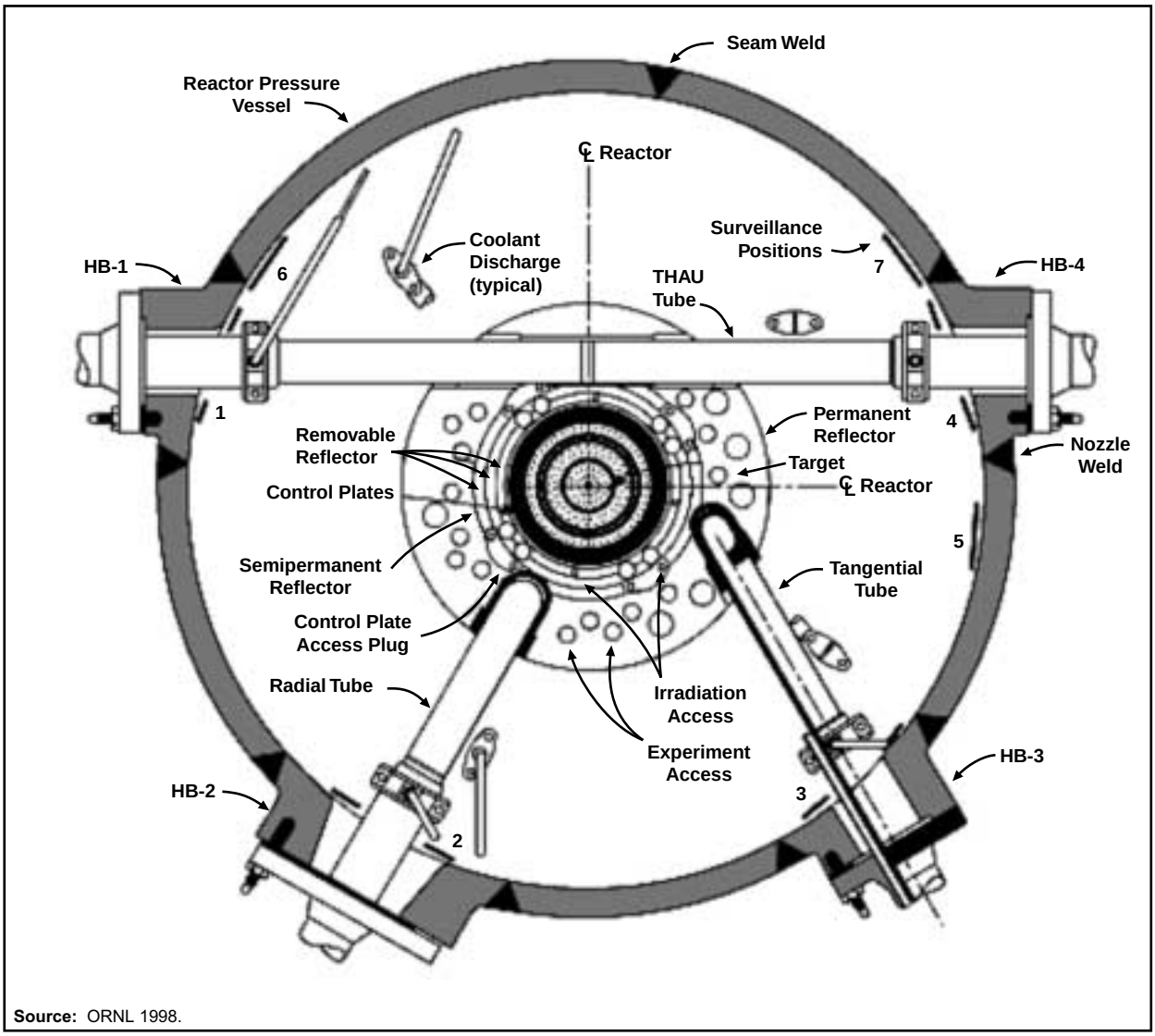
The reactor core assembly is contained in a 2.44-meter (8-foot) diameter pressure vessel in a pool of water. The top of the pressure vessel is 5.18 meters (17 feet) below the pool surface, and the reactor horizontal midplane is 8.38 meters (27.5 feet) below the pool surface. The control plate drive mechanisms are in a subpile room beneath the pressure vessel. These features provide the necessary shielding for working above the reactor core and greatly facilitate access to the pressure vessel, core, and reflector regions.

The neutron flux within HFIR is primarily a thermal neutron flux ranging from approximately 2×10^{15} neutrons per square centimeter per second in the flux trap to approximately 4×10^{14} neutrons per square centimeter per second in the outer regions of the beryllium reflector. Specially designed neutron beam tubes provide access to neutrons that supply intense neutron beams to various specialized instruments used for neutron scattering research.

ORNL produces a variety of medical isotopes using HFIR for irradiation and various hot cell and glovebox facilities for target fabrication and final product purification. **Table 2–2** presents a listing of HFIR-produced therapeutic radioisotopes. Key examples of the therapeutic radioisotopes currently produced in HFIR for distribution include dysprosium-166, rhenium-186, tin-117m, and tungsten-188 (parent of rhenium-188). The nine hydraulic tube positions in the central high flux region permit the insertion and removal of targets at any time during the operating cycle (22 to 24 days) and have traditionally been a major site for the production of medical radioisotopes.

In addition to providing radioisotopes for extramural research and development and commercial applications by distribution through the DOE Isotope Production and Distribution Program, there are medical radioisotope research and development programs at ORNL that depend on the availability of HFIR-produced radioisotopes.

The Isotopes Program at ORNL is totally funded by the DOE Office of Nuclear Energy, Science and Technology's Isotope Production and Distribution Program. It provides enriched stable isotopes, selected radioisotopes, and related technical services for use in a wide variety of research, industrial, and especially medical applications. The scope of work not only includes the production of radioisotopes, but also the development of new methods and equipment to produce, recover, and purify isotope products.



Source: ORNL 1998.

Figure 2–8 Plan View (Cross Section) of HFIR

Table 2–2 Examples of HFIR-Produced Radioisotopes of Current Interest for Therapy

Radioisotope	Half-Life	Target	Comment
Palladium-103	17 days	Palladium-102	Therapy of prostatic carcinoma
Rhenium-186	3.77 days	Rhenium-185	Therapy of prostatic carcinoma
Samarium-153	1.93 days	Samarium-152	Antibodies/bone pain palliation
Tin-117m	13.6 days	Tin-116 or tin-117	Bone pain palliation
Arsenic-77 (from germanium-77)	1.62 days	Germanium-76	Bone pain palliation
Gold-199 (from platinum-199)	3.14 days	Platinum-198	Phosphorus analogue
Tungsten-188 (rhenium-188 daughter)	69 days	Tungsten-186	Bone pain/antibodies/ synovectomy
Dysprosium-166 (holmium-166 daughter)	3.4 days	Dysprosium-164	Synovectomy/bone pain

2.3.1.4 Commercial Light Water Reactor

A CLWR would continue to operate and meet its primary mission requirement, providing steam for the generation of electrical power, in the No Action Alternative, Alternative 1 (FFTF Restart), Alternative 3 (Construct New Accelerator[s]), Alternative 4 (Construct New Research Reactor), Alternative 5 (Permanently Deactivate FFTF [with No New Missions]), and Alternative 2 (Use Only Existing Operational Facilities) when it is not providing irradiation services in support of the plutonium-238 production mission. When the CLWR is supporting the plutonium-238 production mission, it would fully support its primary mission. The production planning assumption for the generic CLWR is 5 kilograms (11 pounds) per year of plutonium-238 or 7.5 kilograms (16.5 pounds) per 18-month operating cycle. Thus, the CLWR alone could meet the program goal of up to 5 kilograms (11 pounds) per year and could be used in combination with any one of the three processing facilities for the plutonium-238 production mission. The use of a CLWR for the medical and industrial isotope production mission and the DOE civilian nuclear energy research and development mission was not considered practical, as discussed in Section 2.6.1.

A typical pressurized water reactor core consists of 170 to 200 fuel assemblies arranged in the reactor vessel in an approximately cylindrical pattern. Most pressurized water reactors operating in the United States are licensed to operate at thermal power levels of 2,500 to 3,500 megawatts (8.5×10^9 to 1.2×10^{10} British thermal units per hour) for net station electrical outputs of 800 to 1,200 megawatts-electric (2.7×10^9 to 4.1×10^9 British thermal units per hour).

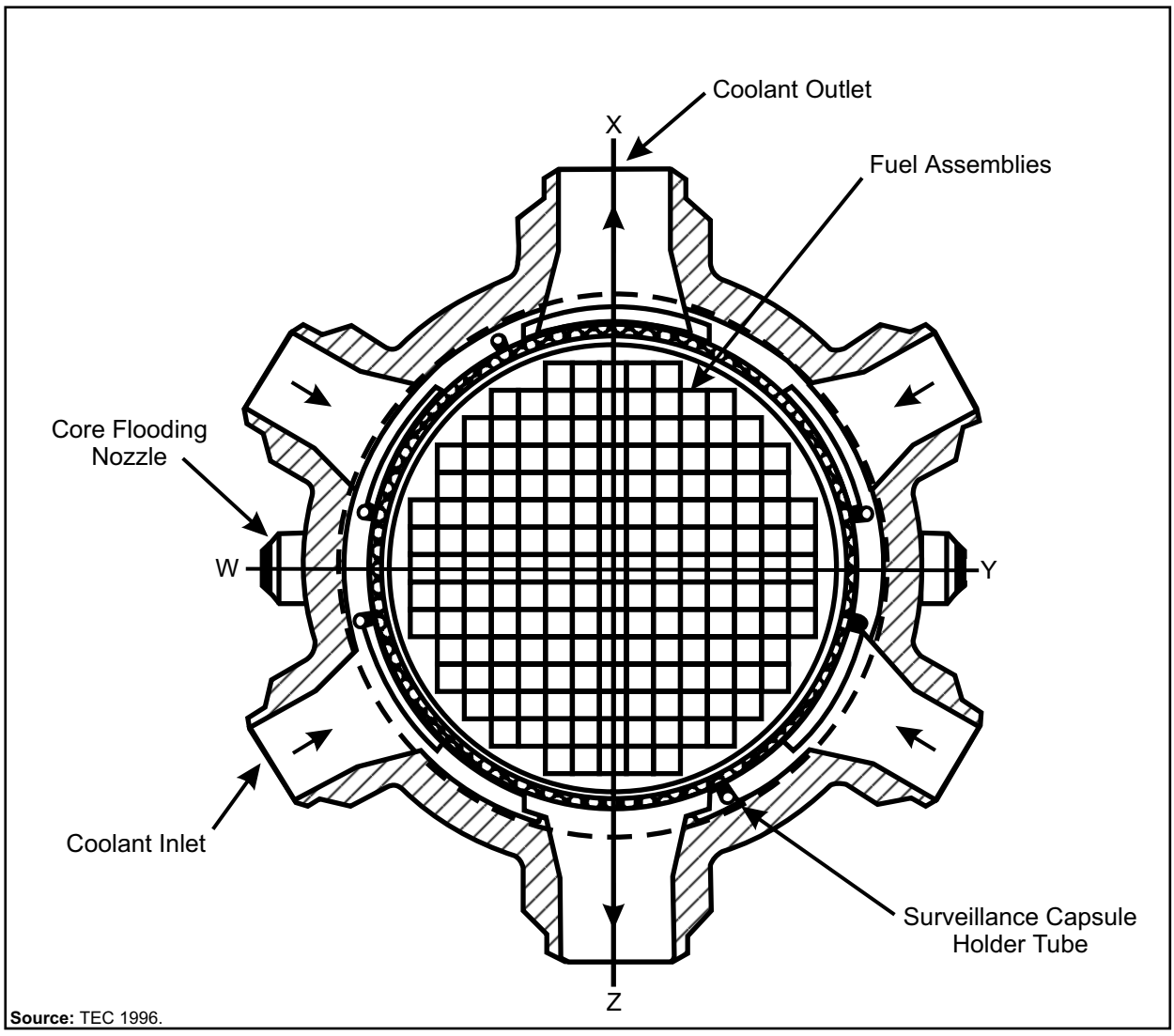
The nuclear steam supply system powered by the pressurized water reactor is generally arranged as two heat transport loops, each with two primary coolant circulating pumps and one steam generator in which the primary coolant dissipates heat generated in the reactor core to the secondary fluid in the steam generator. In addition to serving as a heat transport medium, the primary coolant also serves as a neutron moderator and reflector and as a solvent for the soluble boron used in chemical reactivity control. All nuclear steam supply system components are designed to withstand the effects of earthquakes and loss-of-coolant accidents.

The containment for a pressurized-water reactor plant consists of two structures: (1) a steel containment vessel and (2) a reinforced-concrete shield building.

The containment, including all of its penetrations, is a low-leakage steel structure designed to withstand a postulated loss-of-coolant accident and to confine a postulated release of radioactive material. It houses the reactor pressure vessel, reactor coolant piping, pressurizer, pressurizer quench tank and coolers, reactor primary coolant pumps, steam generators, core flooding tanks, and letdown coolers. Safety systems directly associated with this vessel include the containment spray system, the containment air cooling system, and the containment isolation system. An annular space is provided between the wall of the containment vessel and the shield building. Overhead clearance from the dome of the shield building is also provided.

The shield building itself is a concrete structure surrounding the containment that is designed to provide biological shielding during both normal operations and hypothetical accident conditions. The shield building enables the collection and filtration of fission product leakage from the containment following a hypothetical accident by means of its emergency ventilation system. In addition, the shield building provides environmental protection for the containment from adverse atmospheric conditions and external missiles (e.g., tornado debris).

A complete reactor core of 177 fuel assemblies, arranged in a square lattice that approximates a cylinder, is shown in **Figure 2–9**. All fuel assemblies are identical in mechanical construction and are interchangeable in any core location. The basic fuel assembly (**Figure 2–10**) is normally composed of 208 fuel rods, 16 control rod guide tubes, and one centrally-located position for instrumentation—all within a 15- by 15-position square

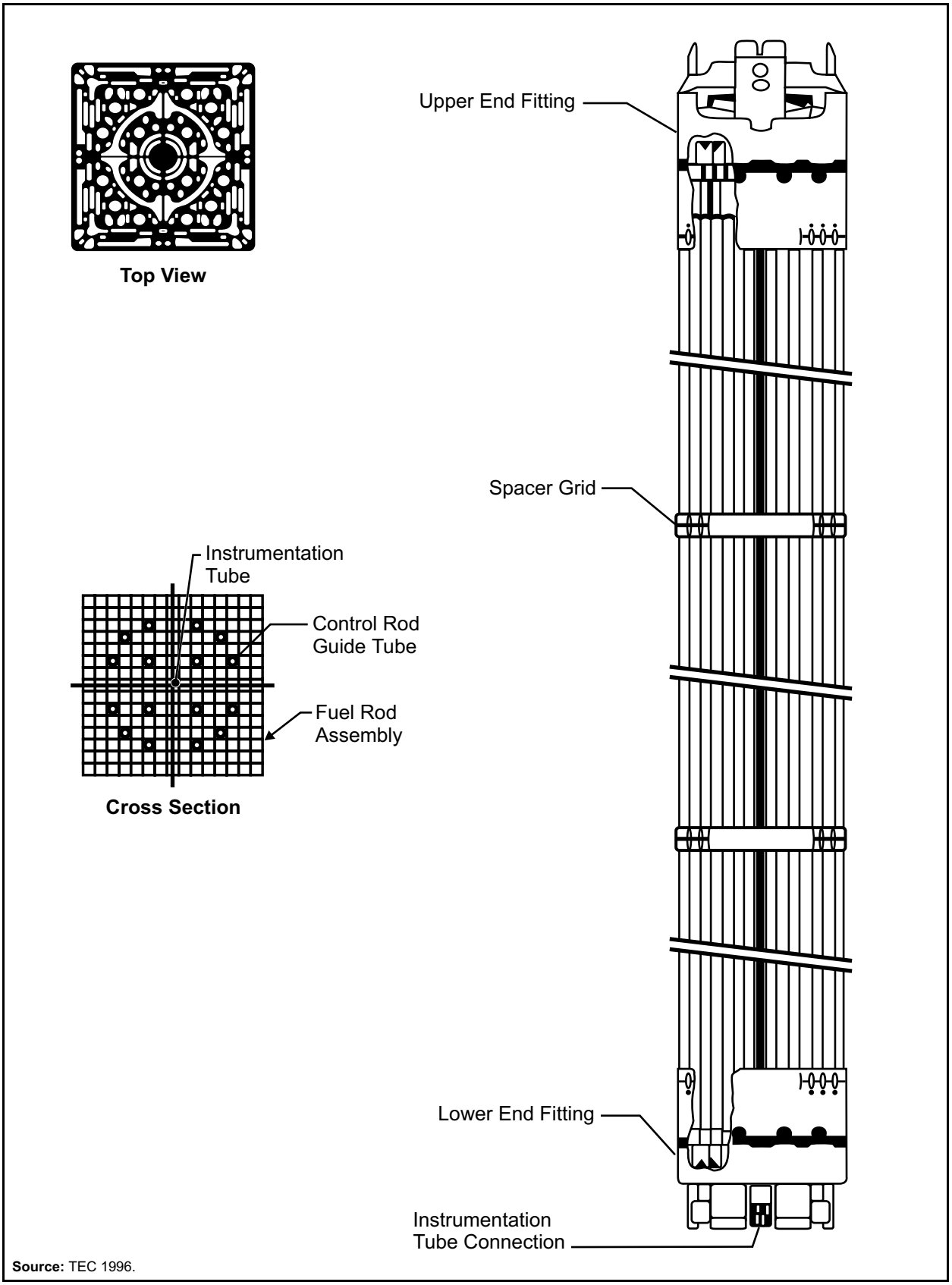


Source: TEC 1996.

Figure 2–9 Plan View (Cross Section) of a Generic CLWR

array. The fuel assembly is approximately 20.3 by 20.3 centimeters (8 by 8 inches) in cross section and has an overall length of 419 centimeters (165 inches).

The neptunium-237 targets can be placed in numerous locations within the reactor core region (i.e., fuel assembly region) and outside of the reactor core region to be irradiated for the production of plutonium-238. Three potential target arrangements were considered for evaluation in this NI PEIS: (1) all targets located in the center fuel assembly position in the reactor core, (2) all targets distributed within the reactor core region, and (3) all targets distributed outside the reactor core region. The center fuel assembly position was selected for evaluation in this NI PEIS because it was assumed that this would be the worst-case location during postulated beyond-design-basis accident conditions. This assumption conservatively postulated that during a beyond-design-basis core disruptive accident, temperatures in the center fuel assembly position would reach levels that would fail the cladding on all the neptunium-237 targets located in that position, resulting in worst-case releases.



Source: TEC 1996.

Figure 2–10 CLWR Fuel Assembly

The substitution of target rods for fuel rod positions in the center fuel assembly would only minimally impact reactor operations. The fuel rods located in the center fuel assembly position normally would not be fresh fuel (i.e., fuel inserted within the first 18-month operating cycle in the reactor); instead, they would be in their second or third operating cycle. The normal power distribution within the core and reactor coolant flow and its distribution within the core would remain within existing technical specification limits.

2.3.1.5 New Accelerator(s)

One or two new accelerators would be constructed and operated in Alternative 3 (Construct New Accelerator[s]). Preconceptual designs have been developed for a low-energy accelerator and a high-energy accelerator for evaluation in this NI PEIS (TechSource 2000). The low-energy accelerator would support the medical and industrial isotope production missions and the civilian nuclear energy research and development mission. This could effectively be accomplished with accelerator energies in the range of 30 to 70 million electron volts. The high-energy accelerator design would support the plutonium-238 production mission and the civilian nuclear energy research and development mission. An accelerator with an energy level of 1,000 million electron volts is required to support the plutonium-238 and nuclear research and development missions. The preconceptual design of the high-energy accelerator presented in Appendix F focused on supporting the plutonium-238 production mission. Although not analyzed in this NI PEIS, the design of the high-energy accelerator could be refined and expanded to perform additional missions such as the production of a select set of medical and industrial radioisotopes. In addition, DOE is aware of longer-term concepts that would apply high-energy accelerators to produce "tuneable" neutrons in a subcritical assembly. Such a facility could be used to address some of the missions more familiar to reactor facilities and may hold considerable promise for future science and technology research. A facility of this nature could provide unique capabilities in areas such as the testing of many different nuclear system coolant, fuel, and materials interactions.

The accelerators would be constructed and operated at one or two existing DOE sites. The low-energy accelerator would be located on the same DOE site as the new support facility or at a DOE site with an existing support facility. The high-energy accelerator could be located at a different DOE site. Alternative 3 site selection was not evaluated as part of this NI PEIS. Because Alternative 3 was evaluated at a generic DOE site, no credit was taken for any existing support infrastructure at the site, and it was postulated that a new support facility would be required to support operation of the low-energy accelerator and its missions and the high-energy accelerator civilian nuclear energy research and development missions if both accelerators were located on the same site. While this approach bounds the environmental impact assessment for the implementation of Alternative 3, it overstates the impacts because this NI PEIS integrates the impacts associated with constructing new support facilities and infrastructure that may be available at the existing DOE site. In the event that Alternative 3 or the low-energy accelerator alone is selected in the Record of Decision for subsequent consideration, follow-on NEPA reviews would evaluate potential locations for either both or one of the accelerators. It is unlikely that DOE would consider locating the new low-energy or high-energy accelerator on a DOE site that does not have an existing infrastructure capable of supporting all or most of the mission requirements. To determine the environmental impacts if Alternative 3 were implemented at a site with adequate support infrastructure, the environmental impacts for the construction of the support facility could be subtracted from the environmental impacts of Alternative 3 as presented in this NI PEIS. Section 4.5 of this NI PEIS presents the environmental impacts from construction and operation of the new support facility separately.

2.3.1.5.1 Low-Energy Accelerator

Three low-energy accelerator options would be available for the production of medical and industrial isotopes and to support nuclear energy research and development: (1) a high-current proton linear accelerator, (2) a

multiparticle cyclotron, or (3) a proton-only cyclotron. The proton-only cyclotron would have distinct technical advantages over the other two options and is described further in the sections that follow.

The proton-only cyclotron can be either a positive proton or negative ion type and is referred to as a proton cyclotron H^+ or proton cyclotron H^-. A positive proton cyclotron would offer lower vacuum requirements and, with the latest technology, a high-extraction efficiency. Obtaining variable energy output would be complicated, however, because extraction can be done using only a single port and splitting the beam would require a complicated septum magnet. In comparison, the negative ion cyclotron would offer a continuous beam with high-current capacity using very simple high-efficiency extraction, a simple method to vary the particle energy, and the possibility of simultaneous irradiation of two different target arrays at different energies. The high-extraction efficiency would be achieved simply by passing the negatively charged beam through a thin foil that strips the electrons from the ion, creating a positive proton. The proton would be directly ejected from the machine by the existing magnetic field with high efficiency (greater than 98 percent). This feature would be important to minimize the activation of the cyclotron structure and thus reduce radiation exposure to the operational staff.

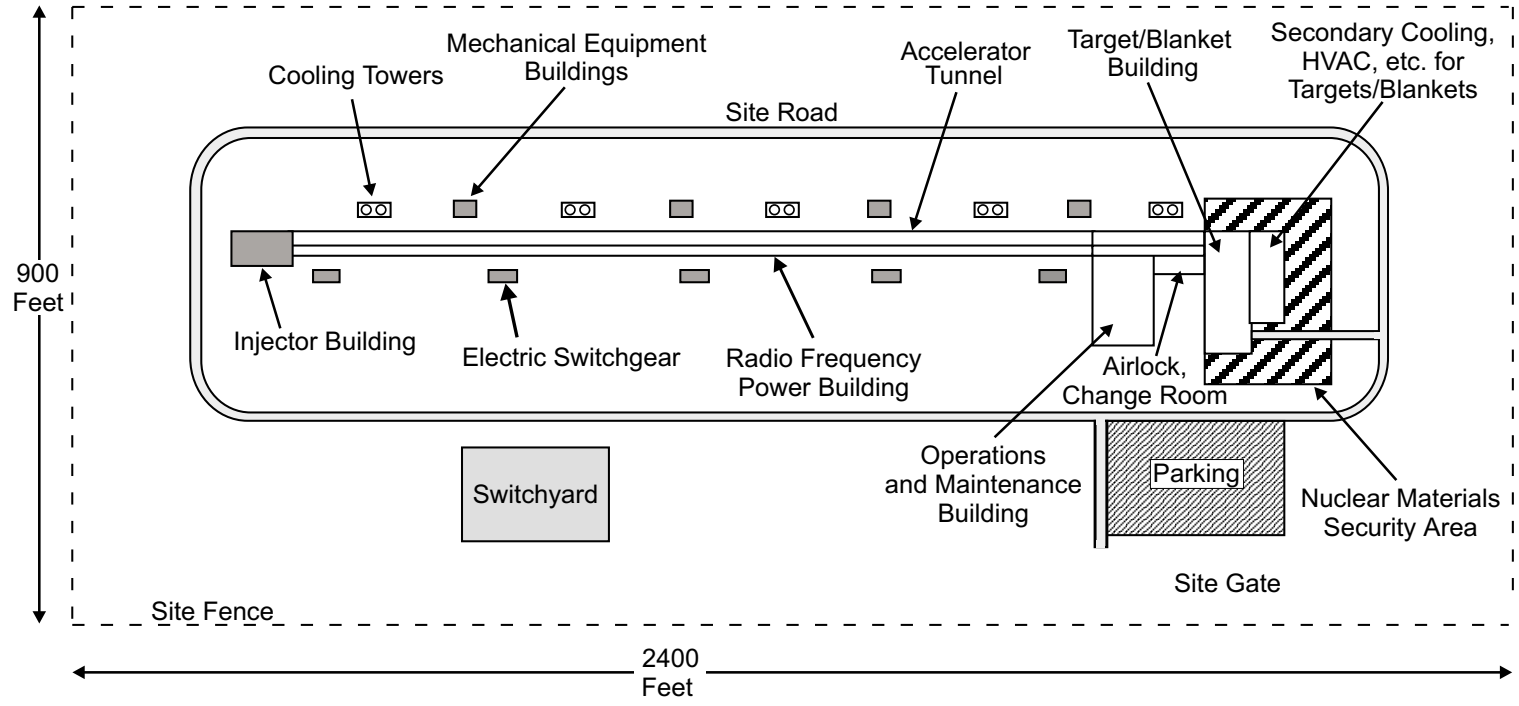
A high-beam current would be advantageous because more products could be prepared in a shorter time. In addition, a much higher specific-activity radioisotope could be prepared at the higher-beam current of the cyclotron. Specific activity is often a critical parameter in many nuclear medicine applications, including research and clinical use. The cyclotron could continuously tune the beam energy, which would be an advantage for research. The ability to tune the energy with precision could also help achieve high-purity isotope production by avoiding energies where impurity isotopes would be readily coproduced. These are important advantages for flexibility in research isotope production and are within the capabilities of commercially proven technology.

A new building with a 43-meter (140-foot) by 43-meter (140-foot) footprint would be constructed to house the cyclotron and the four beam lines. The walls of the facility would be 4.6 meters (15 feet) thick behind the target stations to minimize the neutron flux outside the building. The walls surrounding the cyclotron itself would be 3 meters (10 feet) thick. The mazes throughout the building would have walls 1.5 meters (5 feet) thick so that the total thickness surrounding the cyclotron area would be 3 meters (10 feet). The beam would be diverted to the four target stations by switching magnets located in the cyclotron vault. The beam would be directed through focusing and steering magnets to the target. In the isotope production beam line (northwest cave), the targets would be installed and removed vertically from a hot cell, which would be located on the second floor directly above the target station. The power supplies for the magnets would be housed with the power supplies for the cyclotron. The mechanical equipment for cooling water would be housed in a shielded mechanical room adjacent to the cyclotron vault. Recirculating water for cooling the targets and systems that could contain potentially radioactive material would be separated to prevent cross-contamination. These systems would be contained in mechanical equipment rooms near the respective target station. Piping would be contained in waterproof trenches with leak detection.

See Appendix F for additional details.

2.3.1.5.2 High-Energy Accelerator

In accelerator production of plutonium-238, an energetic beam of protons generated by a linear accelerator would be transported to a heavy metal target where spallation neutrons would be produced and moderated in a surrounding blanket. The blanket containing neptunium-237 would capture the slowed neutrons to produce plutonium-238 through the same nuclear sequence that occurs in a reactor. The accelerator would be housed in a concrete tunnel and buried below ground to provide radiation shielding for operating personnel. **Figure 2–11** presents the layout of the accelerator. A building to house radio frequency power systems and



HVAC Heating, Ventilating, and Air Conditioning

Source: TechSource 2000.

Figure 2–11 Accelerator Production of Plutonium Plant Layout

other equipment used to drive, monitor, and control the accelerator would be above ground close to the accelerator tunnel. The target and blanket assembly would be housed inside a steel and concrete shield within a multistory building that would contain appropriate service equipment. At the target, the small-diameter proton beam transported magnetically from the accelerator would be converted to a much larger cross section by a beam expander to reduce the power density to acceptable levels for the target cooling systems. A source of neutrons produced by an accelerator can be used to produce plutonium-238 from neptunium-237 feedstock through the capture-and-decay nuclear processes. A 1,000-million-electron-volt proton beam produced by a radio frequency linear accelerator would bombard a heavy metal (uranium-238) target, with each proton producing about 40 neutrons. A very preliminary target and blanket design has been developed for scoping purposes, based on the architecture employed in the accelerator production of tritium target and blanket design. It would use uranium-238 (cooled by heavy water) as the neutron-production target. The target would be surrounded by a blanket of neptunium-237 in a dilute mixture of aluminum and water coolant. Enclosing the blanket would be a beryllium reflector.

To meet the plutonium-238 production goal of up to 5 kilograms (11 pounds) per year, the high-energy accelerator facility would conduct three 4-month production campaigns. Each campaign would be divided into 100 days of production and 21 days for recycling the production blanket. A 90 percent plant availability during the scheduled operating periods is assumed. Based on operating experience at the Los Alamos Neutron Science Center Linear Accelerator, 90 percent plant availability would be achievable. See Appendix F for additional details.

The preconceptual design of the high-energy accelerator presented in Appendix F focused on supporting the plutonium-238 production mission. While not evaluated in this NI PEIS, the design of the high-energy accelerator could be refined and expanded to perform additional missions such as the production of a select set of medical and industrial radioisotopes. In addition, DOE is aware of longer-term concepts that would apply high-energy accelerators to produce “tuneable” neutrons in a subcritical assembly. Such a facility could be used to address some of the missions more familiar to reactor facilities and may hold considerable promise for future science and technology research. A facility of this nature could provide unique capabilities in areas such as the testing of many different nuclear system coolant, fuel, and materials interactions. The accelerator designs for Alternative 3 were developed to a level of detail that was adequate to assess the environmental impacts associated with the construction and operation of the proposed facilities and the technical feasibility of meeting the mission objectives. In the event that the NI PEIS Record of Decision selects Alternative 3, DOE would prepare conceptual, preliminary, and detailed designs and optimize the facility designs to accomplish the stated missions. Additional NEPA review would be required for site selection and to evaluate the environmental impacts of integrating the more refined accelerator designs with the existing site infrastructure(s).

2.3.1.6 New Research Reactor

A new research reactor would be constructed and operated in Alternative 4 (Construct New Research Reactor). A preconceptual design for a new research reactor was developed to meet the following DOE missions: (1) producing medical and industrial isotopes, (2) producing plutonium-238 (annual production of up to 5 kilograms [11 pounds]), and (3) supporting nuclear energy research and development. In accordance with U.S. nuclear nonproliferation policy, a design limitation of this new research reactor would be that it could only use low-enriched uranium with an enrichment of less than 20 percent uranium-235. This preconceptual design includes the basic elements of the research reactor facility, which are sufficient to support this NI PEIS, but does not include the design details (e.g., system and layout drawings, bill of materials, electrical and piping routing) commensurate with a complete preliminary reactor design.

The reactor design was developed to a level of detail that was adequate to assess the environmental impacts associated with the construction and operation of the proposed facilities and the technical feasibility of meeting the mission objectives. The design of the new research reactor is based on current research reactor designs that have been approved by both NRC and the International Atomic Energy Agency, as well as the nuclear regulatory authorities of many nations. Reactor core physics calculations were performed to evaluate three different nuclear fuel designs (described in Appendix E). Based on this analysis, the desired mission for this reactor, current nuclear fuel manufacturing capabilities, and safety considerations, a TRIGA (training, research, isotopes General Atomics) production reactor fuel design was selected for the new research reactor. The principal distinguishing features of the TRIGA fuel are its proven safety performance during power pulsing and its demonstrated long-term irradiation integrity.

To concurrently produce medical and industrial isotopes, meet the plutonium-238 production goal of up to 5 kilograms (11 pounds) per year, and provide irradiation services for civilian nuclear energy research and development, it was determined that a reactor core power of 50 megawatts-thermal would be necessary. Higher power levels and alternative target designs capable of meeting production requirements were also considered in the new research reactor design analysis but were not analyzed in this NI PEIS. For example, although not analyzed in this NI PEIS, operating at 100 megawatts-thermal could reduce the amount of neptunium-237 required to meet the plutonium-238 production requirements.

At the 50-megawatts-thermal power level, the core would require an active cooling system with forced coolant flow to maintain the fuel below its material thermal limits. The new research reactor cooling system would use a tank within a pool that is connected to primary coolant circulating pumps, heat exchangers, and an ultimate heat sink consisting of two cooling towers. The pool would be housed in a reactor building that also would enclose the pumps, heat exchangers, secondary systems, and spent nuclear fuel storage pool. The spent nuclear fuel storage pool, sized to store the reactor core's discharged spent nuclear fuel for its entire 35-year production period, could be hydraulically connected to the reactor core pool for refueling and emergency reflooding. The ultimate heat sink cooling towers, air exhaust stack, and emergency diesel generators would be located outside the reactor building.

The fuel for the new research reactor would be based on an extension of currently licensed low-enriched uranium TRIGA fuel designs for 10- to 16-megawatts-thermal reactors. The new research reactor fuel design would be identical to current low-enriched uranium TRIGA fuel for higher power cores, except the new reactor fuel would have a larger assembly configuration array (i.e., 8 by 8 versus 4 by 4) and a longer active fuel length (153.7 centimeters [60.5 inches] versus 55.88 centimeters [22.0 inches]). The larger array and length were selected to meet the plutonium-238 production requirements and to maintain high safety factors with respect to fuel thermal performance.

Along with the fuel rods, the core would contain a number of medical and industrial isotope and plutonium-238 production target rods. These rods would occupy positions in a fuel assembly where a fuel rod would otherwise exist. Each of these positions would have an Incoloy-800 alloy guide tube with the same dimensions as the fuel rod cladding. The target rods would be inserted into these guide tubes for their design irradiation time period. In addition, some fuel rod positions in core fuel assemblies would be replaced with similar guide tubes to accommodate Incoloy-800-clad boron carbide control rods. Boron carbide is a widely used, proven, and accepted neutron absorber for control rods. **Figure 2–12** presents a representative illustration of the fuel rod; the neptunium-237, medical, or industrial radioisotope target rod; and the control rod. **Figure 2–13** shows a cross-sectional view of each type of fuel assembly in the core. The new research reactor core design would consist of 68 fuel assemblies, each of which would be enclosed in a square aluminum shroud for structural support and coolant flow control. Key design features of the core are provided in Appendix E.

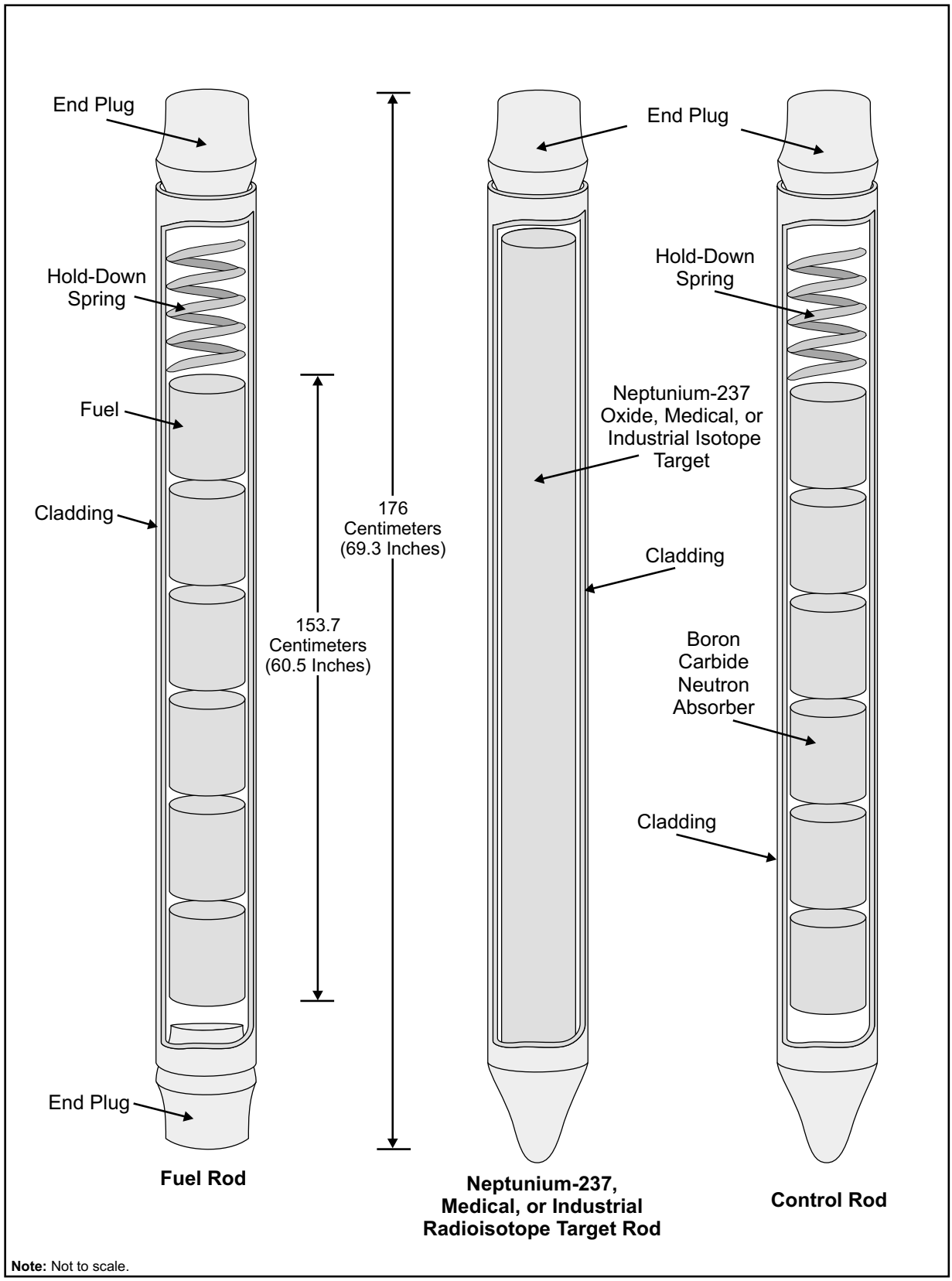


Figure 2–12 Representative Illustration of Fuel Rod; Neptunium-237, Medical, or Industrial Radioisotope Target Rod; and Control Rod (New Research Reactor)

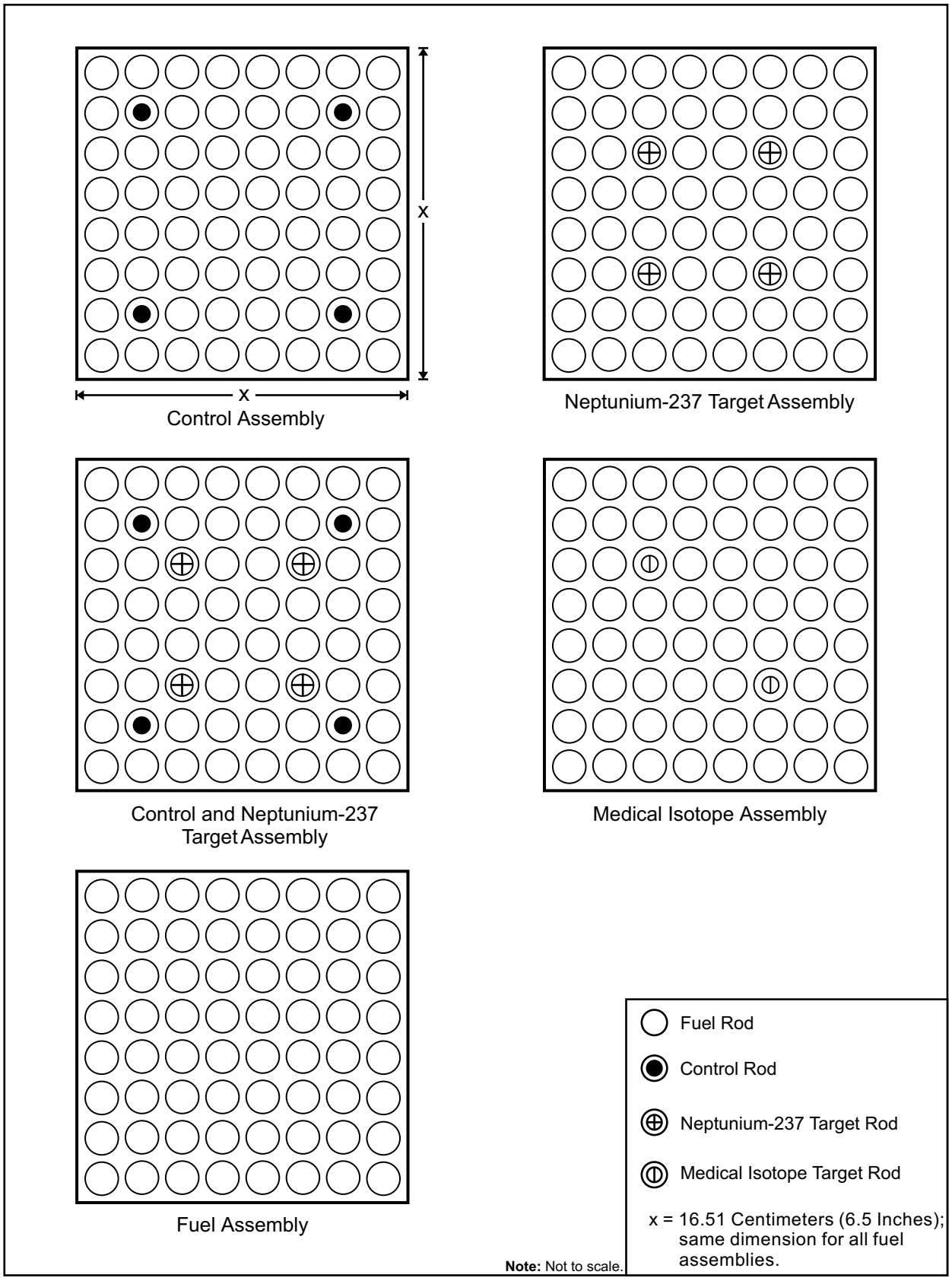


Figure 2–13 Cross-Sectional View of Fuel Assemblies in the Core (New Research Reactor)

The core would include eight rabbit tubes for short irradiation time production of medical or industrial isotopes and civilian nuclear energy research and development. These rabbit tubes would be located outside the fuel region of the core, but still within an area with a relatively high neutron flux. A cross-sectional view of the new research reactor core showing the layout of fuel assemblies, target rod assemblies, control rod assemblies, reflector, and rabbit tubes is presented in **Figure 2–14.**

The new research reactor would be constructed and operated at an existing DOE site. Since the potential site has not been selected, it is evaluated in this NI PEIS as a generic DOE site. Because Alternative 4 was evaluated at a generic DOE site, no credit was taken for any existing support infrastructure at the site, and it was postulated that a new support facility would be required to support operation of the new research reactor and its medical isotope production and civilian nuclear energy research and development missions. While this approach bounds the environmental impact assessment for the implementation of Alternative 4, it overstates the impacts because this NI PEIS integrates the impacts associated with constructing new support facilities and infrastructure that may be available at the existing DOE site. In the event that Alternative 4 is selected in the Record of Decision for subsequent consideration, follow-on NEPA reviews would evaluate potential site locations. It is unlikely that DOE would consider locating the new research reactor on a DOE site that does not have an existing infrastructure capable of supporting all or most of the mission requirements. To determine the environmental impacts if Alternative 4 were implemented at a site with adequate support infrastructure, the environmental impacts for the construction of the support facility could be subtracted from the environmental impacts of Alternative 4 as presented in this NI PEIS. Section 4.6 of this NI PEIS presents the environmental impacts from construction and operation of the new support facility separately.

Reactor Operation

Operation of the new research reactor would be similar to other research reactors except that the core would be maintained at full power for a minimum 80 percent of the year. At the beginning of a cycle of operation, neptunium-237 and medical isotope target rod assemblies that require a long irradiation time would be inserted into their appropriate fuel assembly sleeve locations. The target rods would be mechanically attached to a cluster spider assembly similar to that used for the control rod assembly. The neptunium-237 target rod assemblies would remain in the core for the entire annual fuel cycle. These target rod assemblies would be removed from the host fuel assembly without removing the fuel assembly from the core, and then would be transferred to the spent fuel storage pool using the transfer canal. Medical and industrial isotope target rods that require a 100-day irradiation cycle would be removed and replaced with new target rod assemblies during brief reactor shutdown periods. These target rod assemblies would be removed and transferred in a manner similar to that of the neptunium-237 target rod assemblies. Isotopes that require only a short irradiation time would be inserted into rabbit tubes for the required 10- to 25-day time period. The eight rabbit tubes would be located outside the core, but inside the reflector region. The insertion and removal of irradiation targets in the rabbit tubes would have no significant effect on core reactivity and would not affect power operation.

After an isotope-specific cooling time in the spent fuel pool, the medical and industrial isotope and neptunium-237 target assemblies would be transferred to a shipping cask in the spent fuel storage pool. Using the overhead crane in the spent fuel pool area, shipping casks would be placed onto a truck in the reactor building bay adjacent to the fuel storage pool for shipment to the processing facility. New targets would be shipped from the target preparation facility into the reactor building bay by truck, transferred into the spent fuel storage pool, and subsequently moved to the reactor core pool or rabbit tube area for insertion into the core.

The plutonium-238 annual production goal of up to 5 kilograms (11 pounds) was calculated to be achieved with a 300-day annual irradiation time, which corresponds to a capacity factor of approximately 80 percent. Key reactor annual resource requirements are presented in Appendix E.

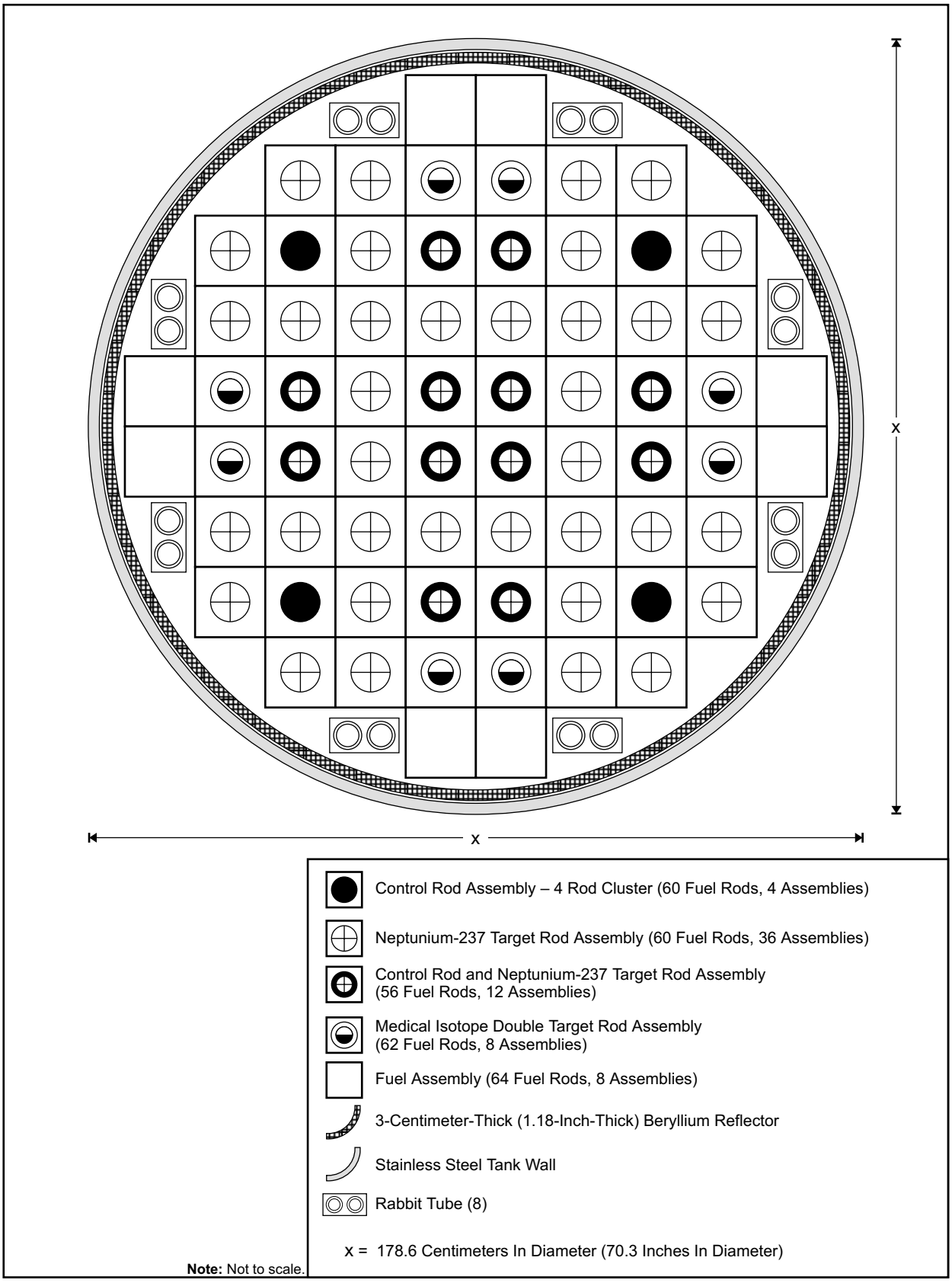


Figure 2–14 Cross-Sectional View of Research Reactor Core

Reactor Construction

Construction of the new research reactor facility was determined to require 4 years after design and licensing activities have been completed (AECL 1996; ANSTO 1999).

2.3.2 Target Fabrication and Postirradiation Processing Facilities

The proposed DOE facilities that would be used for the fabrication, storage, and postirradiation processing of the targets necessary for the program mission are (1) REDC at ORNL, (2) FDPF and/or Building CPP–651 at INEEL, (3) FMEF at Hanford, (4) RPL/Building 306–E at Hanford, or (5) a new target fabrication and processing facility at an existing DOE site that would support medical and industrial isotope production for targets irradiated in the proposed new low-energy accelerator or research reactor facilities. REDC, FDPF, and CPP–651 would support plutonium-238 production; FMEF would support both plutonium-238 and medical and industrial isotope production. The RPL/306–E facilities and the new facility would support only medical and industrial isotope production.

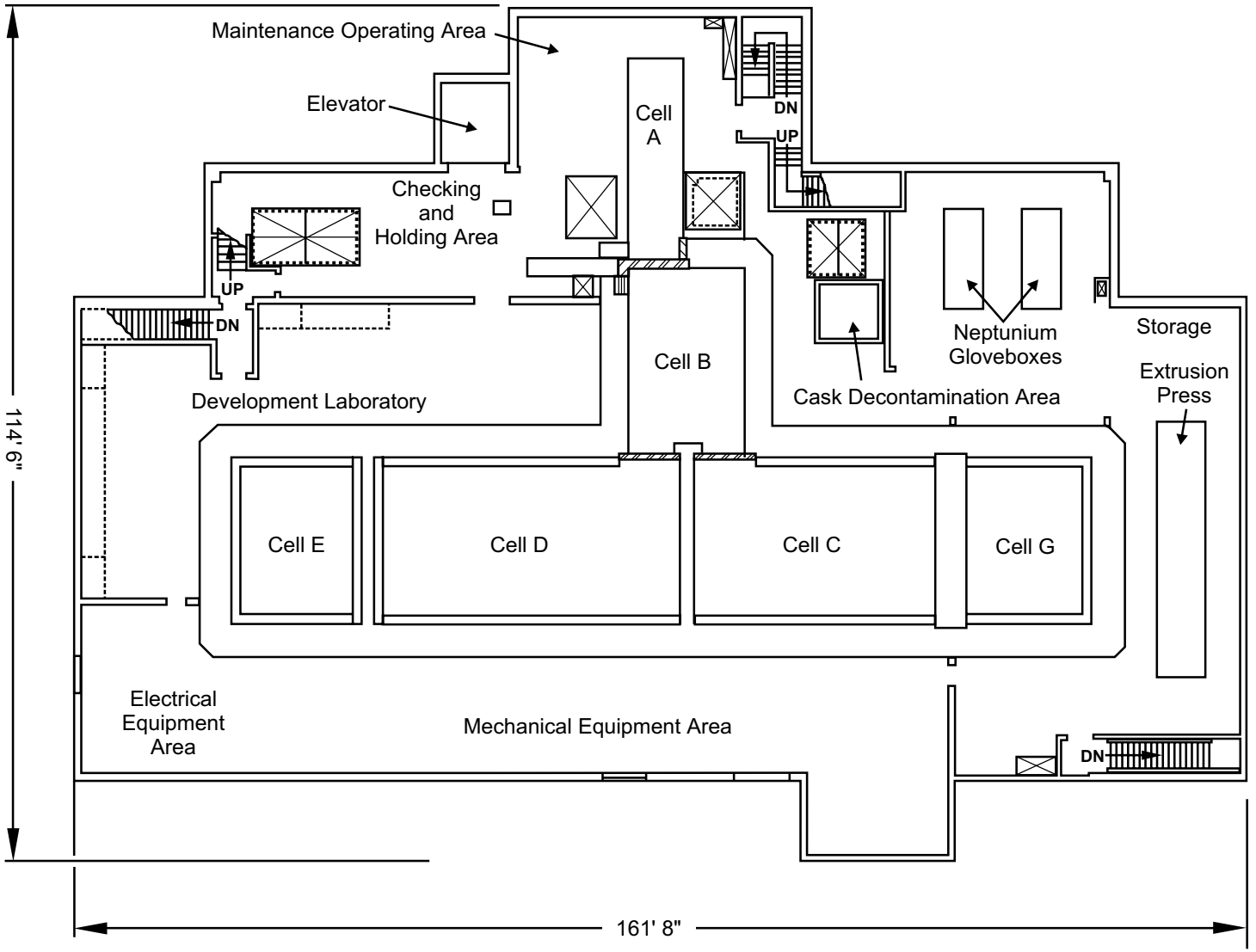
2.3.2.1 Radiochemical Engineering Development Center

REDC at ORNL is a companion facility to HFIR. The REDC's two buildings house heavily shielded hot cells and analytical laboratories that are used for remote fabrication of rods and targets (for irradiation in HFIR) and processing of irradiated rods and targets for the separation and purification of transuranic elements, process development, and product purification and packaging.

ORNL's REDC Building 7930 is proposed for storage of neptunium-237 under one option of the No Action Alternative. It also is proposed for storage of neptunium-237, fabrication of neptunium-237 targets, and processing of irradiated neptunium-237 targets under two irradiation options in Alternative 1 (FFTF Restart), three irradiation options in Alternative 2 (Use Only Existing Operational Facilities), and one irradiation option each in Alternative 3 (Construct New Accelerator[s]) and Alternative 4 (Construct New Research Reactor). REDC's current radiochemical missions would not be impacted by the addition of the proposed storage of neptunium-237, fabrication of neptunium-237 targets, and the processing of irradiated neptunium-237 targets activities. REDC would have no role in support of Alternative 5 (Permanently Deactivate FFTF [with No New Missions]). Figure 3–1 presents a map of ORR that depicts REDC's location.

REDC Building 7930 is divided into four major areas: (1) a cell complex with seven cells, six shielded and one unshielded; (2) maintenance and service areas surrounding the cell complex; (3) an operating control area; and (4) an office area adjacent to, but isolated from, the operating areas. Utility services, ventilating systems, crane and manipulator systems, and liquid waste systems also are included. The proposed plutonium-238 processing and storage activities would require equipment installation in three main areas of the second floor of REDC Building 7930. A plan view of these areas is shown in **Figure 2–15.** REDC hot cell facilities that would be used for the proposed action have never been used. The activities required for target fabrication would take place in shielded gloveboxes. The mechanical operations involved in the final target fabrication process may present lesser hazards that permit them to be carried out in open boxes. Cell E would contain processing equipment to purify the separated plutonium-238 product, prepare the plutonium oxide, and transfer the oxide into shipping containers. Cell E would also contain vertical storage wells for dry storage of neptunium and other actinides.

Cell D activities would include receipt of irradiated targets, as well as target dissolution, chemical separation of neptunium and plutonium from fission products, and partitioning and purification of neptunium. Cell D



Note: To convert feet to meters, multiply by 0.3048.
Source: Wham et al. 1998.

Figure 2–15 REDC Building 7930 Second Floor Plan View (Cross Section)

also contains process equipment for removing transuranic elements from the aqueous waste streams and vitrifying the waste.

The neptunium dioxide (NpO_2) containers would be stored in specially designed storage vaults to provide secure, safe storage for the materials. DOE safeguards and security guidelines would be followed whenever the material is being stored, transported, or processed. Detailed descriptions of the facility and the processes associated with storage, target fabrication, and postirradiation processing are provided in Appendix A.

2.3.2.2 Fluorinel Dissolution Process Facility

FDPF is in the Idaho Nuclear Technology and Engineering Center (INTEC) that is northeast of the Central Facilities Area at INEEL and approximately 3.2 kilometers (2 miles) southeast of ATR. FDPF is proposed for fabrication of neptunium-237 targets and processing of irradiated neptunium-237 targets under two irradiation options in Alternative 1 (FFTF Restart), three irradiation options in Alternative 2 (Use Only Existing Operational Facilities), and one irradiation option each in Alternative 3 (Construct New Accelerator[s]) and Alternative 4 (Construct New Research Reactor). Figure 3–6 presents a map of INEEL that depicts FDPF's location.

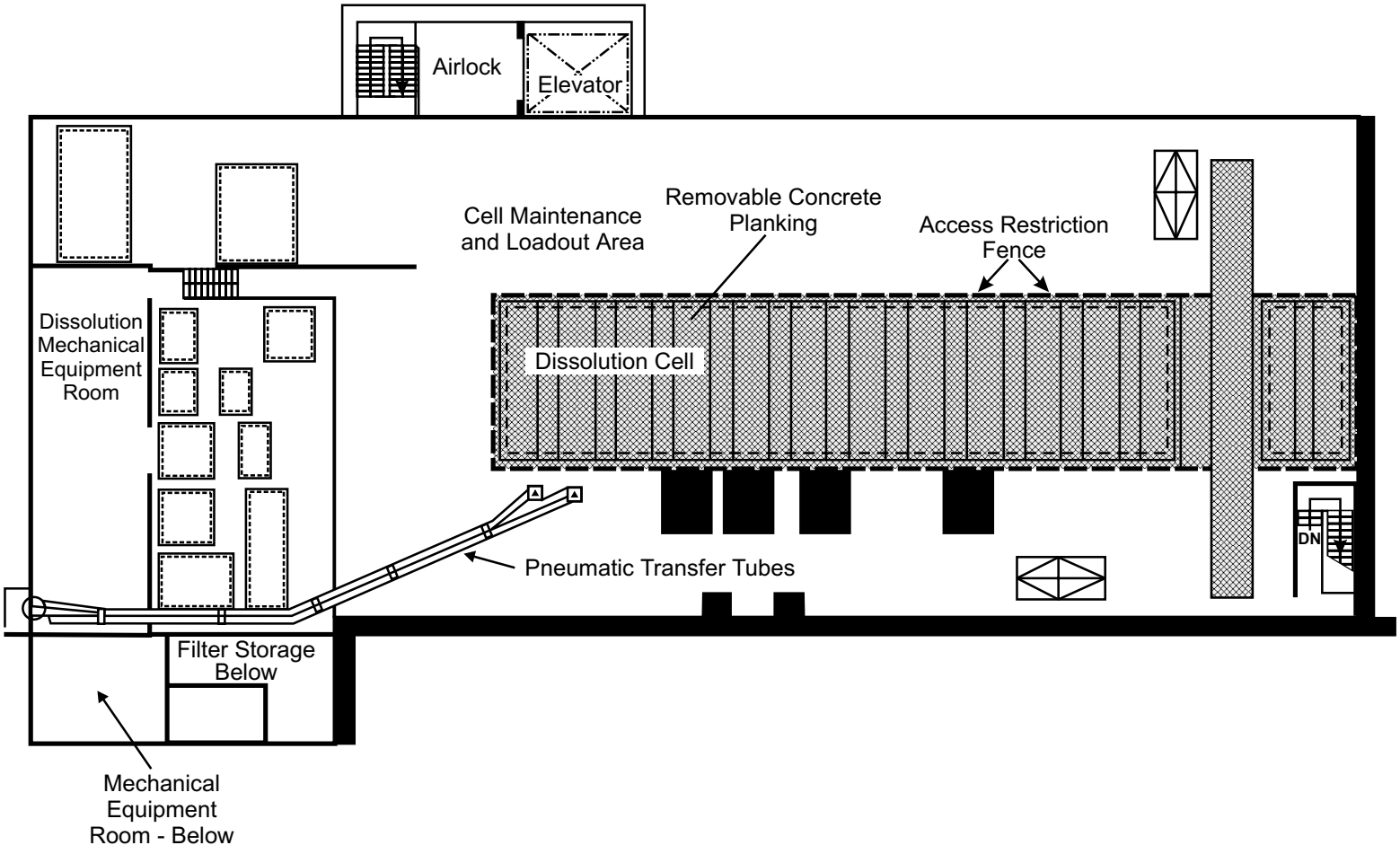
FDPF has no current mission. Historically, INTEC reprocessed spent nuclear fuel from U.S. Government reactors to recover reusable highly enriched uranium. After DOE announced in April 1992 that it would no longer reprocess spent fuel, reprocessing operations at INTEC ended. Two buildings at INTEC are candidate storage and processing sites for plutonium-238 production: Building CPP–651, the Unirradiated Fuel Storage Facility, and Building CPP–666, FDPF. Building CPP–651 was originally designed for the storage of special nuclear materials to support Defense Programs and is quite flexible in terms of the size and shape of special nuclear materials that it can receive and store. The 100 storage positions in the vault use the existing structural barriers of Building CPP–651 (earth and concrete) and provide supplemental security protection via their in-ground concrete storage silo design. Each storage position houses a rack that holds seven highly enriched uranium product cans. Racks are raised and lowered in their storage positions via an overhead 1-ton hoist.

Building CPP–666 is divided into two parts, the Fuel Storage Facility and FDPF. The Fuel Storage Facility consists of receiving and unloading areas, a fuel unloading pool, and six storage pools for storing nuclear fuel.

FDPF was designed and built to process Navy fuel via three dissolver trains. When fuel reprocessing was discontinued, uranium and hazardous materials were flushed from FDPF, and the facility is currently under consideration for new missions. FDPF consists of a large hot cell and supporting areas with a total area of approximately 3,700 square meters (40,000 square feet). The facility is divided into five levels that are identified by their elevation relative to ground level (Hochhalter 1982). A floor plan of the plus 28-foot level, the proposed location for target fabrication activities, is shown in **Figure 2–16**.

The chemical separation would take place in the FDPF cell using small centrifugal contactors installed for that purpose. Storage of neptunium-237 would be performed in Building CPP–651, which is located within 100 meters (328 feet) of FDPF. There are 100 in-ground concrete-shielded storage well positions in this vault. Each storage well contains a rack that can be modified to house cans of neptunium-237.

The neptunium dioxide containers would be stored in specially designed storage vaults to provide secure, safe storage for the materials. DOE safeguards and security guidelines would be followed whenever the material is being stored, transported, or processed.



Source: Hochhalter 1982.

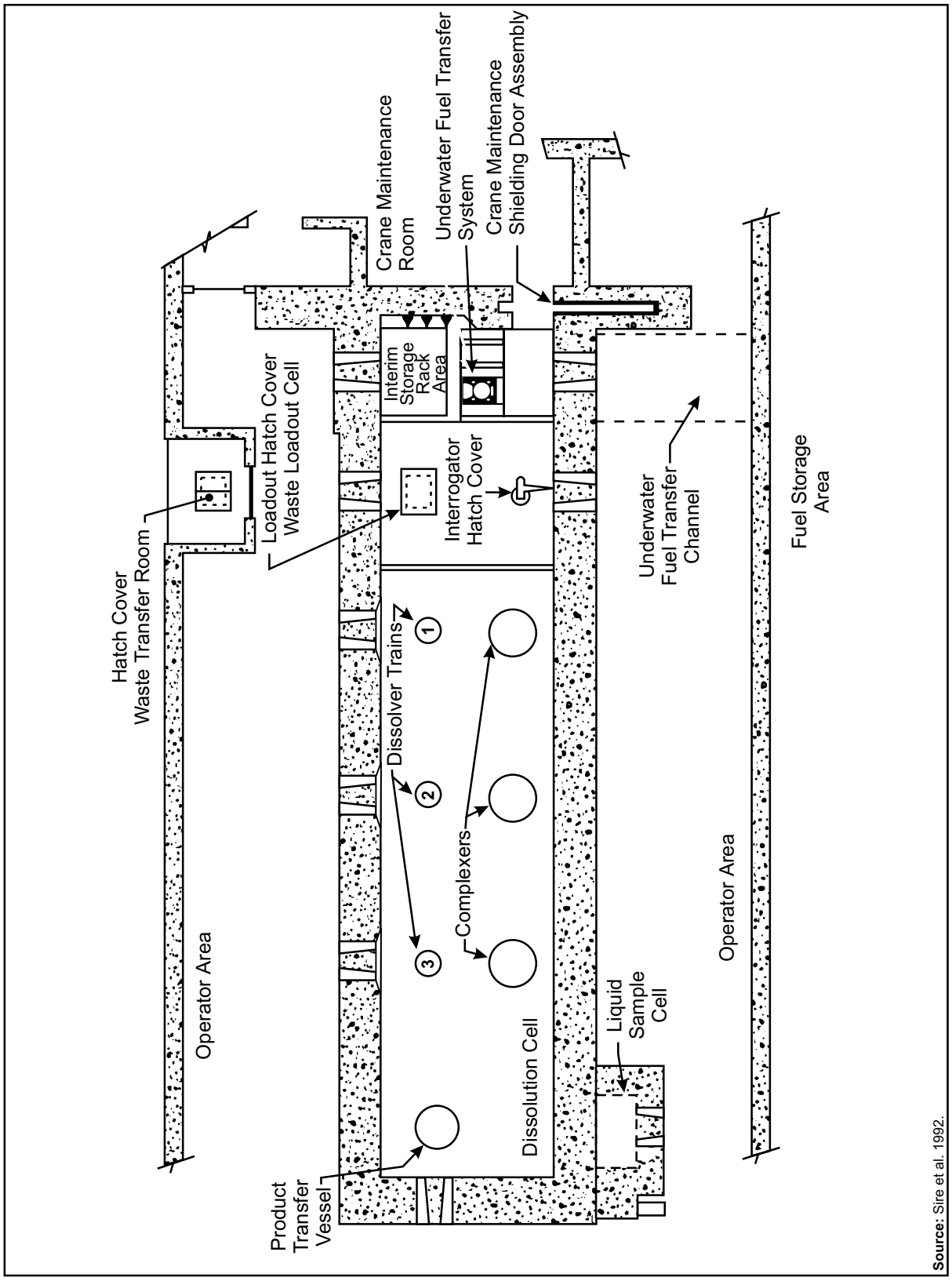
Figure 2–16 Fluorinel Dissolution Process Facility, Plus 28-Foot Level

The FDPF portion of INTEC includes a hot cell about 6.1 meters (20 feet) wide, 30.5 meters (100 feet) long, and 15.2 meters (50 feet) deep that is shielded by 1.8-meter-thick (6-foot-thick) concrete walls, as shown in **Figure 2–17**. The cell was designed to process Navy spent nuclear fuel via three dissolver trains, each of which consists of a 1,700-liter (450-gallon) Hastelloy C-4 dissolver and a 6,510-liter (1,720-gallon) Hastelloy C-4 complexer vessel in series. Each train is connected to a common 8,000-liter (2,110-gallon) stainless steel product transfer vessel that was used for accountability sampling prior to transferring the adjusted fuel dissolution product for solvent extraction separations. If the targets were dissolved in a continuous process, a small, 12.5-liter (3.3-gallon) dissolver would be skid-mounted on the grate at the level of the dissolver lids, and the dissolvers would be used for collecting the dissolution product of irradiated neptunium-237 targets. If a batch dissolution process were used, a small 200-liter (53-gallon) dissolver system designed for the small target dissolution throughput rate could be skid-mounted on the grate at the level of the dissolver lids for batch processing. Three complexer vessels in the cell could be used for waste or rework solution collection, or for the collection of condensate if a waste evaporator were employed. The head-end dissolution system is supported by remote manipulators (overhead and master/slave), as well as an underwater fuel transfer system and crane for target transfer and waste loadout. The dissolver offgas system scrubs potentially hazardous chemicals and filters radioactive particles from the offgases of the process vessels before releasing them into the heating, ventilating, and air conditioning exhaust system. The chemical separation would take place in the FDPF cell using small centrifugal contactors installed for that purpose. The storage of neptunium-237 would be performed in either FDPF or in a secure vault facility, Building CPP–651, located within 100 meters (328 feet) of FDPF. There are 100 in-ground concrete-shielded storage well positions in this vault. Each storage well contains a rack that can be modified to house cans of neptunium-237. Detailed descriptions of the facility and the processes associated with storage, target fabrication, and postirradiation processing are provided in Appendix A.

2.3.2.3 Fuels and Materials Examination Facility

Use of Hanford's FMEF is proposed for storage of neptunium-237 under one option of the No Action Alternative. It also is proposed for storage of neptunium-237, fabrication of neptunium-237 targets, and processing of irradiated neptunium-237 targets under two irradiation options in Alternative 1 (FFTF Restart), three irradiation options in Alternative 2 (Use Only Existing Operational Facilities), and one irradiation option each in Alternative 3 (Construct New Accelerator[s]) and Alternative 4 (Construct New Research Reactor). In addition to the support of the plutonium-238 production mission activities in Alternative 1, FMEF would also support medical and industrial production mission and civilian nuclear energy research and development mission activities at Hanford. FMEF would have no role in supporting Alternative 5 (Permanently Deactivate FFTF [with No New Missions]). FMEF is adjacent to the west of FFTF in the 400 Area of Hanford. Figure 3–12 presents a map of Hanford that depicts FMEF's location.

FMEF was built during the late 1970s and early 1980s as a major addition to the breeder reactor technology development program at Hanford. Although it has never been used, the facility was constructed to perform fuel fabrication and development and postirradiation examination of breeder reactor fuels (DOE 1995b). FMEF is currently being maintained in a condition suitable for a future mission. In 1998, FMEF was placed into a partial layup condition to reduce the cost of maintaining the facility. Many systems were shut down and most hazardous materials were removed from the building. FMEF is considered clean and uncontaminated because no nuclear materials have been introduced (Hoyt et al. 1999). Some critical systems, such as the fire detection and protection systems, remain in operation. In order to avoid freezing of the fire protection water systems, limited heating and ventilating remain available. For example, the heating, ventilating, and air conditioning system has been modified to simplify its operation by blocking automatic dampers in appropriate configurations. Also, although the chillers have been laid up, including removal of the refrigerant, the chilled water system (containing an ethylene glycol-water mixture) remains available to help distribute heat within the building. Electrical power and lighting remain available, and the freight elevator remains in service to



Source: Sire et al. 1992.

Figure 2–17 Plan View of FDPF Cell

support routine facility walkdowns and any required maintenance. FFTF staff conduct surveillance and maintenance of FMEF.

FMEF consists of a 30-meter-high (98-foot-high) Process Building that has an attached Mechanical Equipment Wing on the west side and an Entry Wing on the south (front) side. The Mechanical Equipment Wing houses utility and support equipment, including water treatment equipment, air compressors, and a portion of the air conditioning equipment.

The Entry Wing contains space for reactor fuel assembly (recently used as a training facility in support of the Hanford Site cleanup mission), lunchroom and change rooms, and heating and air conditioning equipment associated with the Entry Wing. Personnel access into the Process Building is provided via a Security Guard Station and automated personnel access control portals located on the first floor of the entry wing. Office space and administrative support areas are also housed on the second floor of the Entry Wing (DOE 1995b).

The Process Building is approximately 53.3 meters (175 feet) wide by 82.3 meters (270 feet) long and extends from around 10.7 meters (35 feet) below grade to 30 meters (98 feet) above grade. Total potential operating space is approximately 17,470 square meters (188,000 square feet). The Process Building contains several large interconnected hot cells and many smaller connected hot cells. Major cranes are available, but some cranes, windows, and manipulators were not installed because construction of FMEF was halted prior to completing work on the hot cell complex (Hoyt et al. 1999). Nevertheless, the building is divided into six operating floors or levels that are identified in the following manner by their elevation relative to ground level and their primary function:

- The top floor at the 21.3-meter (70-foot) elevation is called the Secure Automated Fabrication Level. This level contains the Secure Automated Fabrication Line, automated fabrication equipment originally designed to produce reactor fuel.

- The lower Fuel Fabrication Level at the 13-meter (42.5-foot) elevation consists of two separate operating areas—one designated as the Low Gamma Test Pin Fabrication and Development Area and the other as the Unit Process Cell. This level provides approximately 470 square meters (5,100 square feet) of potential operating space around the Unit Process Cell. This cell area is highly shielded by thick concrete walls and was intended for the future development of remote fabrication and maintenance equipment or for the production of high gamma test pins. However, this cell area is not equipped at this time.

- The lower Chemistry Level at the 6.5-meter (21.25-foot) elevation surrounds the upper portions of the Nondestructive Examination Cell and the Decontamination Cell, which extend upward from the floor below. This level was designed to contain equipment to perform the chemical analyses of fuel material necessary to support fuel fabrication work. Much of the work planned in this area was to be performed in gloveboxes to reduce personnel radiation exposures. Also located on this level is an automated system that is potentially available for handling and storing the special nuclear material, such as the feed material for the fuel fabrication processes. The area encompasses approximately 790 square meters (8,500 square feet) of potential operating space.

- The Entry Level at ground level is the main operating floor of the Nondestructive Examination Cell, which also extends into the floors above and below. The Nondestructive Examination Cell was designed to contain remotely operated equipment for the nondestructive examination of irradiation fuel assemblies and pins. Maintenance and decontamination of equipment were to be performed in the adjacent Decontamination Cell. The Entry Level also contains computer and operations control rooms and inert gas systems and building air exhaust equipment. The Shipping and Receiving Area,

which is approximately 500 square meters (5,400 square feet) of operating floor space, is at the extreme east end of the Process Building on the Entry Level. This area includes a liquid waste loadout station, a solid waste storage area, a truck lock, and a large high-bay material-handling area.

- The Equipment Level at the minus 5.3-meter (17.5-foot) elevation was designed to contain a variety of support equipment, including two separate electrical switchgear rooms, emergency air compressors, heating and ventilating system air supply equipment, Nondestructive Examination Cell inert atmosphere equipment, emergency batteries, analytical chemistry cell exhaust equipment, and building air filtering system components. Also included is the vacuum equipment associated with the vacuum and air sample vacuum systems.

- The DE Cell Level at the minus 10.7-meter (35-foot) elevation (see **Figure 2–18**) contains cells originally intended for destructive examination of fuels and materials samples. These cells are arranged in two parallel rows along a horizontal transfer corridor that was to be used to transfer equipment between individual cells. The DE Cell area is heavily shielded, and work in the cells was planned to be performed using remotely operated equipment. The Entry Tunnel extends from below the Shipping and Receiving Area floor (on the Entry Level) to the DE Cell Level (10.8 meters or 35.5 feet total height). The Entry Tunnel was designed to house a 75-ton rail-mounted transporter intended to transfer casks between the Shipping and Receiving hatch and the Decontamination Cell and Nondestructive Examination Cell Floor penetrations. The transporter rails are roughly halfway up the tunnel at the 5-meter (16.2-foot) elevation.

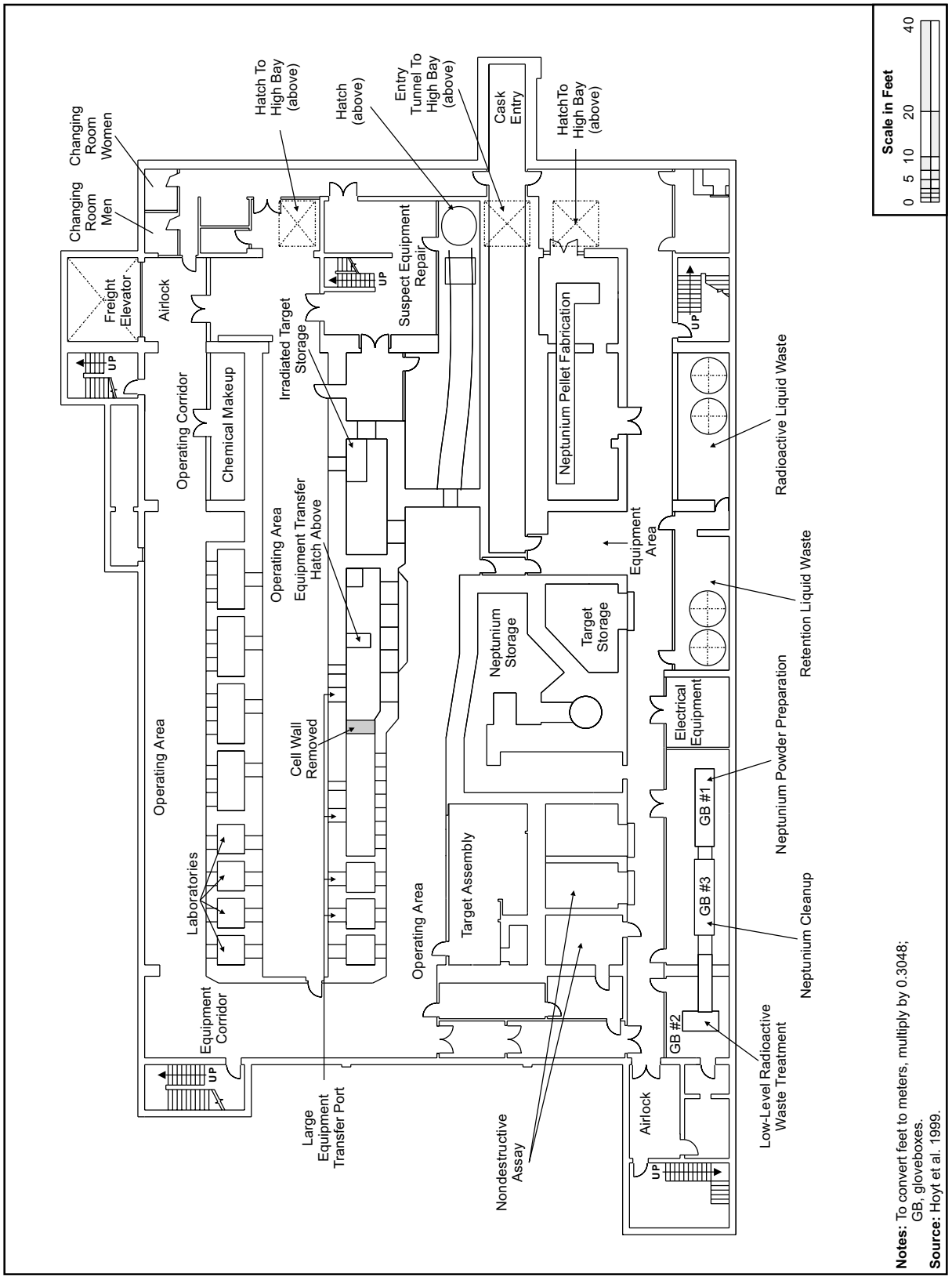
The use of FMEF for neptunium-237 target material storage, target fabrication, and postirradiation processing would require the construction of a new 76-meter (250-foot) stack. The neptunium dioxide (NpO_2) containers would be stored in specially designed storage vaults to provide secure, safe storage for the materials. DOE safeguards and security guidelines would be followed whenever the material is being stored, transported, or processed. Detailed descriptions of the facility and the processes associated with storage, neptunium-237 target fabrication, and postirradiation processing in support of plutonium-238 production are provided in Appendix A.

2.3.2.4 Radiochemical Processing Laboratory and Building 306–E

Two Hanford 300 Area facilities are proposed to support medical and industrial isotope target fabrication and postirradiation: RPL and Building 306–E (RPL/Building 306–E). The facilities support the four irradiation options of Alternative 1 (FFTF Restart) that are not supported by FMEF. RPL and Building 306–E would be used to support medical and industrial isotope production and civilian nuclear energy research and development activities. These activities would not impact current missions at the facilities. RPL and Building 306–E have no role in support of the No Action Alternative, Alternative 2 (Use Only Existing Operational Facilities), Alternative 3 (Construct New Accelerator[s]), Alternative 4 (Construct New Research Reactor), and Alternative 5 (Permanently Deactivate FFTF [with No New Missions]). Figure 3–12 presents a map of Hanford that depicts the locations of RPL and Building 306–E. The following descriptions are based on the *Hanford Data Request for FFTF Operational Support Facilities* (BWHC 1999).

2.3.2.4.1 Radiochemical Processing Laboratory

The research and development activities of the Radiochemical Processing Group are conducted at RPL in the 300 Area of Hanford. RPL consists of a central area that contains general purpose laboratories designed for low-level radioactive work, a front wing that contains office space and shops, and two annexes that provide shielded enclosures with remote manipulators for high-level radiochemical work. The facility also contains



Notes: To convert feet to meters, multiply by 0.3048; GB, gloveboxes.
Source: Hoyt et al. 1999.

Figure 2–18 FMEF Layout, Minus 35-Foot Level

laboratories and specialized facilities designed for work with nonradioactive materials, microgram-to-kilogram quantities of fissionable materials, and up to megacurie quantities of radionuclides. RPL would be the primary site for fabricating the radioactive targets (i.e., targets containing radium-226 or recycled materials from previous irradiations). Total space within RPL is 13,350 square meters (143,700 square feet), of which 4,140 square meters (44,500 square feet) are occupied by general chemistry laboratories. The floor plans for the first floor and the basement of RPL are shown in **Figures 2–19** and **2–20**. A recent space utilization survey of RPL indicated that 646 square meters (6,950 square feet), representing 15.6 percent of the laboratories' area, are presently unoccupied. All of the occupied and nearly all of the unoccupied laboratories are functional and are fully equipped with standard utilities. Several of the laboratories, especially those used for radioanalytical work, have been renovated during the past few years. Upgrading and modernization of the equipment within the chemistry laboratories has been given a high priority during the past 2 years. During the space utilization survey at RPL, an assessment was made of the number of fume hoods and shielded gloveboxes (including several small hot cells) that are available in the chemistry laboratories for additional programmatic work. Of the 79 functional fume hoods and 23 shielded gloveboxes, 50 fume hoods and 15 gloveboxes are available for additional work.

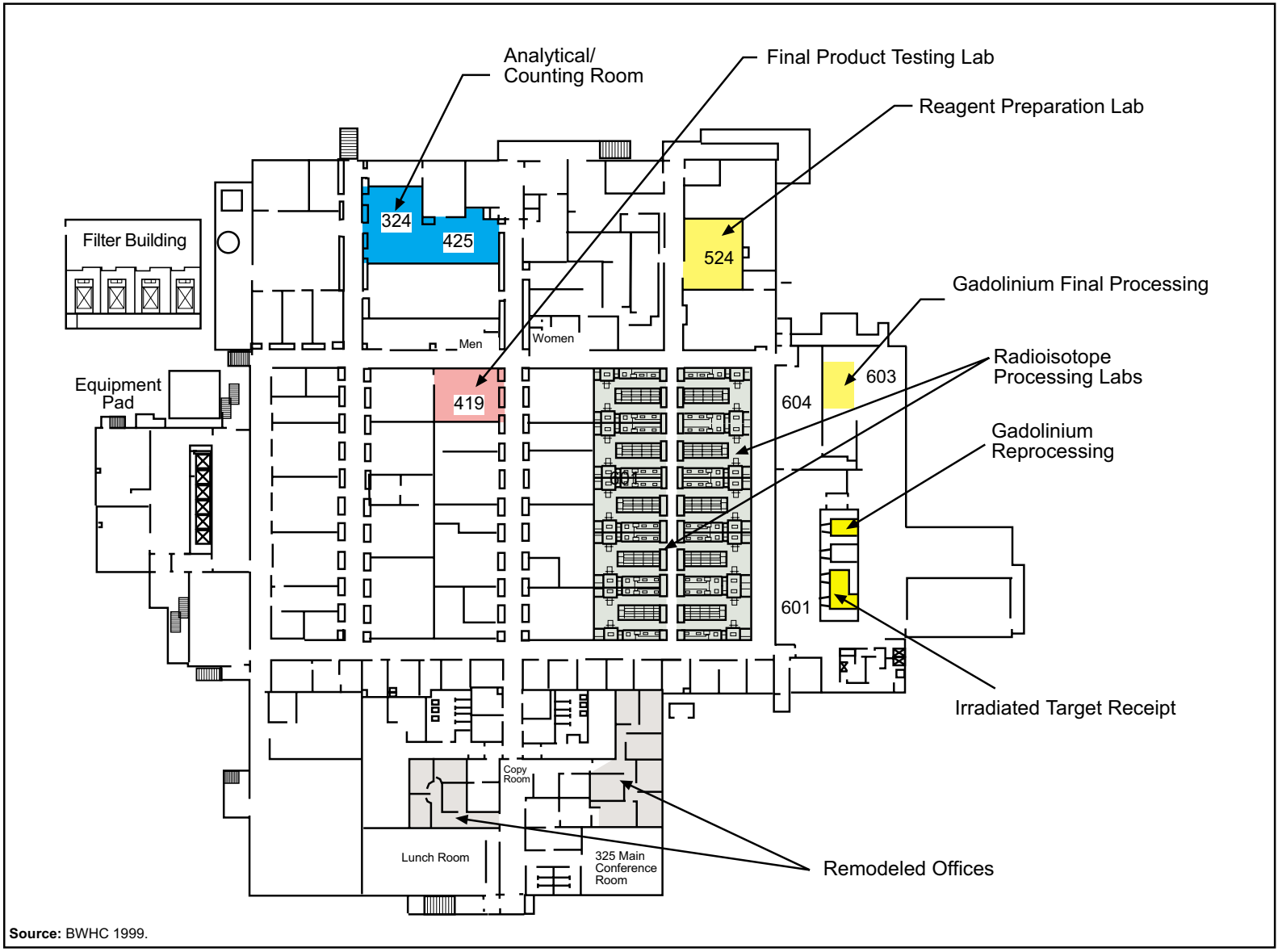
A special feature of RPL is the existence of two heavily shielded hot cell facilities located in annexes on the east and west sides of the building. These shielded facilities are the High-Level Radiochemistry Facility and the Shielded Analytical Laboratory. These two hot cell complexes are heavily used because they provide capabilities for conducting bench-scale to pilot-scale work with a wide variety of highly radioactive materials. Their capabilities include those required to conduct radiochemical separation and purification procedures, irradiated fuel or target sectioning and processing, metallography, physical properties testing of activated metals, thermal processing (including waste vitrification), and radioanalytical and preparatory chemistry operations.

The High-Level Radiochemistry Facility contains three large, interconnected hot cells designated as A-Cell, B-Cell, and C-Cell. Each of the three cells is 4.6 meters (15 feet) high and 2.1 meters (7.0 feet) deep. The A-Cell is 4.6 meters (15 feet) wide, and the B-Cell and C-Cell are each 1.8 meters (6.0 feet) wide. In-cell operations are performed using medium-duty electromechanical manipulators, and operators view their work through leaded-glass, oil-filled windows. Closed-circuit television cameras and videocassette recorders have been installed for detailed inspection work within the hot cells. The A-Cell and C-Cell also have overhead bridges that contain hoists with a 2,200-kilogram (4,840-pound) capacity. The hot cells are fully equipped with utilities and have shielded service penetrations at the front wall to allow insertion of special instruments. Each hot cell contains several process vessels located below the work deck that range in capacity from 4.0 to 320 liters (1.1 to 84.5 gallons). A large shielded door and a shielded double-door transfer port located in the rear wall of the cell provide access to each hot cell in the High-Level Radiochemistry Facility. Cask payloads weighing up to 2,200 kilograms (4,840 pounds) can be transferred into and out of the hot cells using a bridge crane located in the canyon behind the cells.

The Shielded Analytical Laboratory contains six interconnecting hot cells, each of which is 1.7 meters (5.5 feet) wide, 1.7 meters (5.5 feet) deep, and 2.9 meters (9.5 feet) high. Each hot cell is equipped with a pair of medium-duty manipulators. Turntables built into the rear walls of the hot cells provide rapid transfers of radioactive samples into and out of the cells. The Shielded Analytical Laboratory hot cells are equipped to perform a wide variety of analytical chemistry operations with highly radioactive samples.

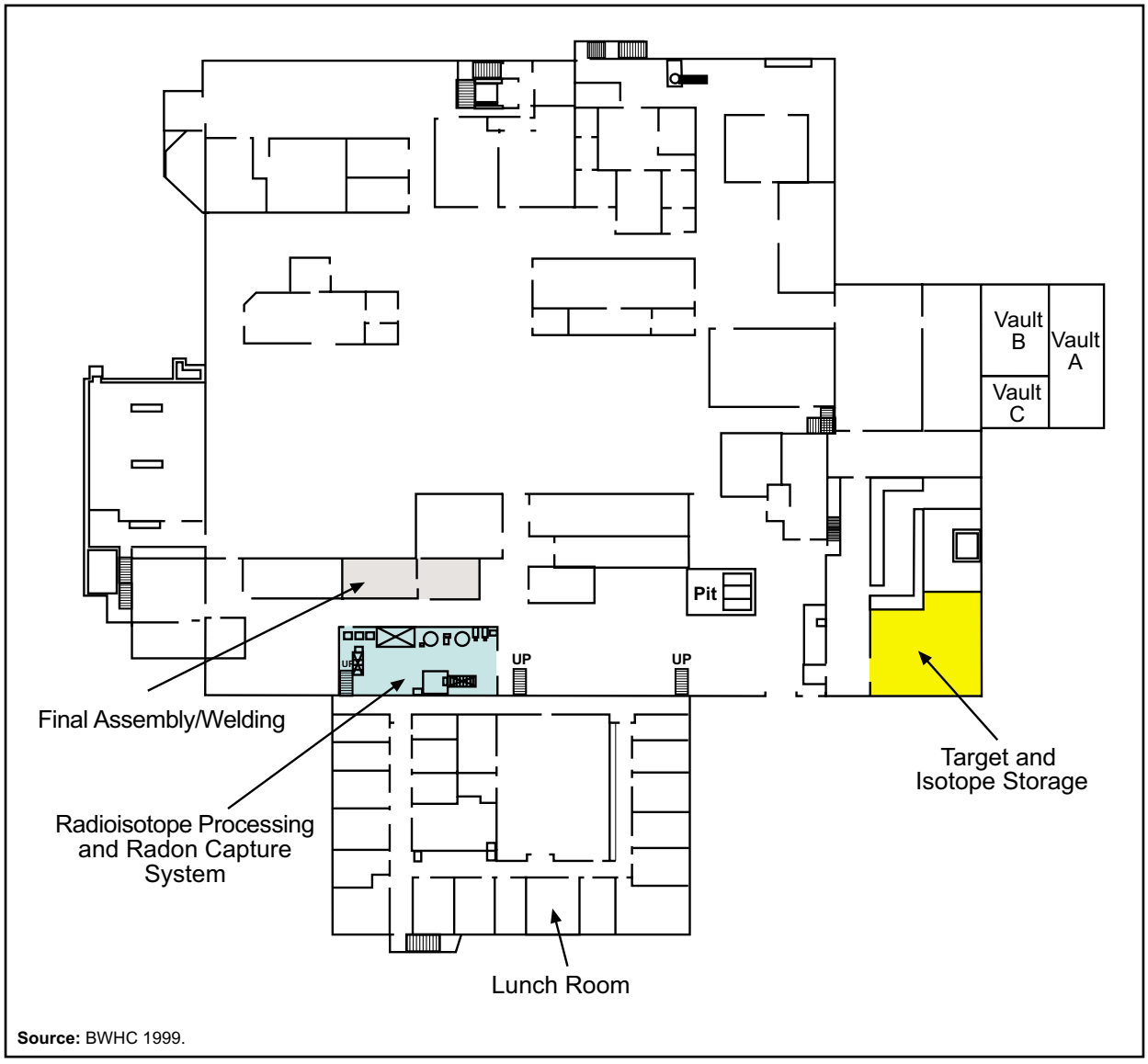
The primary features and functions of the laboratories within RPL that would be used for processing targets irradiated at FFTF are described below.

- A cluster of 10 laboratories would be available on the first floor of RPL. Each laboratory would contain a small hot cell, a shielded glovebox, and a fume hood with interconnecting transfer ports.



Source: BWHC 1999.

Figure 2–19 RPL: Proposed First Floor Locations for Hot Cell Operations and Radiochemical and Radioanalytical Laboratories for FFTF Target Processing



Source: BWHC 1999.

Figure 2–20 RPL: Proposed Basement Locations for Assembly, Processing, and Storage of FFTF Targets; Also Shown: Laboratory with Radon Gas Capture System to be Used for Processing Radium-226 Targets

- A transfer port for receiving casks containing irradiated targets into the A-Cell of the High-Level Radiochemistry Facility would be installed, and provision would be made in the C-cell for initial processing of highly radioactive targets (e.g., irradiated europium targets containing gadolinium-153 product).

- Target preparation and storage areas would be provided in the basement of RPL, in close proximity to the facilities where the radioactive and recycled targets would be assembled and welded.

- A 139.4-square-meter (1,500-square-foot) laboratory equipped with a radon gas capture system would be available in the basement of RPL to process radium-226 targets and the product isotopes generated by irradiation of these targets (all of these targets generate radon gas as intermediate products in their decay chains).

Detailed descriptions of the processes associated with medical and industrial isotope storage, target fabrication, and postirradiation processing are included in Appendix C.

2.3.2.4.2 Building 306–E

Building 306–E was constructed in 1956 as part of the nuclear material production program at Hanford and was used to develop the coextrusion process for N-Reactor fuel. Major upgrades and renovations were completed in the late 1960s and early 1970s to support the civilian reactor development program (Liquid Metal Reactor Program—FFTF). The building has 4,273 square meters (46,000 square feet) of floor space, with a 36.5-meter by 61-meter by 6.4-meter-high (120-foot by 200-foot by 21-foot-high) bay containing one 1.5-ton, one 5-ton, and three 10-ton cranes. The facility has electron beam and laser welding, certified nondestructive testing, a 3.7-meter by 3.7-meter (12-foot by 12-foot) vertical assembly and test station with a 24.4-meter (80-foot) hook height, a machine shop, and an instrument development laboratory. A description of the spaces is provided below; a view of the floor plan is provided in **Figure 2–21**.

Function	Area (square feet)
Offices	4,298
Laboratories	25,003
Shops	2,358
Conference	511
Common	14,133
Total	46,303

The building is serviced by three 1,416-cubic-meters-per-minute (50,000-cubic-feet-per-minute) supply units complete with filters, steam coils, and spray chambers. Two of the units have refrigeration coils for summer cooling. Two ceiling-mounted 1,012-cubic-meters-per-minute (35,750-cubic-feet-per-minute) recirculation fans with freon compressors provide additional cooling and air movement. Fume hoods have individual exhaust fans. Chemical and acid tanks exhaust through two 340-cubic-meters-per-minute (12,000-cubic-feet-per-minute) fume scrubbers to a 12.2-meter-high, 7.6-centimeter-diameter (40-foot-high, 3-inch-diameter) stainless steel exhaust stack. Equipment exhaust collects through a grid that leads to two 566-cubic-meters-per-minute (20,000-cubic-feet-per-minute) exhaust fans. Plastic hoods and duct work are provided for highly corrosive service.

Major equipment includes three industrial x-ray machines, a 6-kilowatt Hamilton Standard electron beam welder, five open face hoods, two inert gas welding chambers, and one electrolytic cutoff saw. Utilities include hot and cold water, deionized water, propane, helium, compressed air, argon, steam, and sanitary and process sewers as well as a special acid drain and neutralizing tank. Normal power is provided by a 1500-kilovolts ampere transformer with 150-kilovolts ampere backup power from an adjoining building and a 30-kilovolts ampere emergency transformer. The building is protected by redundant emergency alarm systems, fire gongs, and an evacuation siren.

2.3.2.5 New Support Facility

A new generic support facility would have the mission of preparing medical and industrial isotope targets for irradiation, processing irradiated targets, and housing the materials research and development activities in association with Alternatives 3 and 4. Siting of the generic support facility for medical and industrial isotope production would require the facility to be located in the same general vicinity (0.2 to 20 kilometers [0.12 to 12.4 miles]) as the new irradiation facility (accelerator or reactor). Colocation with the irradiation facility

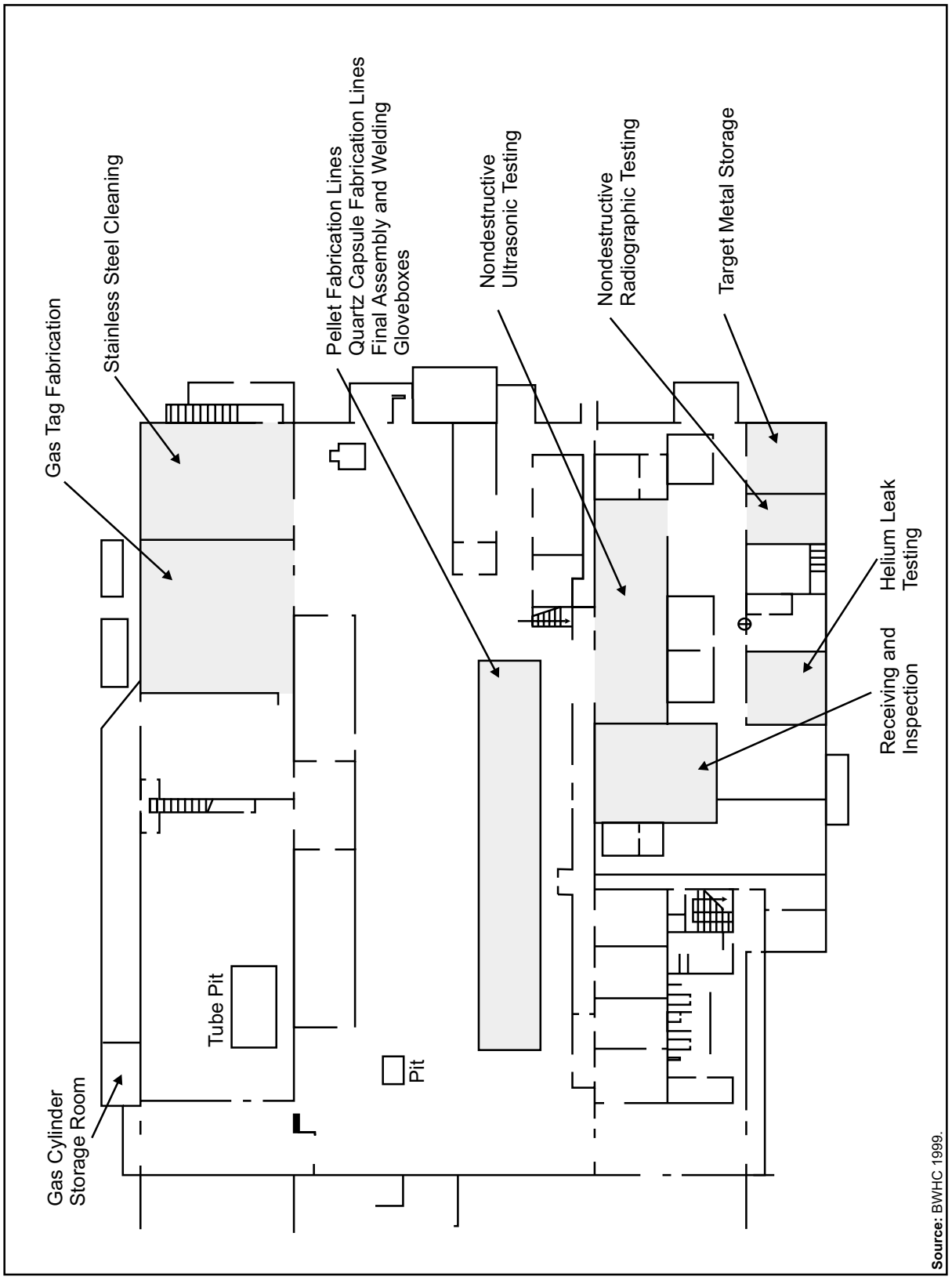


Figure 2–21 Building 306–E Floor Plan

would be needed to process some irradiated target materials promptly after removal from the reactor/accelerator because some isotopes have short half-lives. Colocation would also minimize transportation time. Although the facility could be located within the irradiation facility security protection area, the lack of a defense mission and fissile material in the generic support facility indicates that a high level of physical protection would not be warranted.

The generic support facility mission would be accommodated by a one-story, 3,345-square-meter (36,000-square-foot) above-grade building with a 1,490-square-meter (16,000-square-foot) basement area under a portion of the footprint (SAIC 2000). **Figure 2–22** provides the general layout of the building. The relative position and size of the processing, support, office, and research and development areas are shown on the diagram.

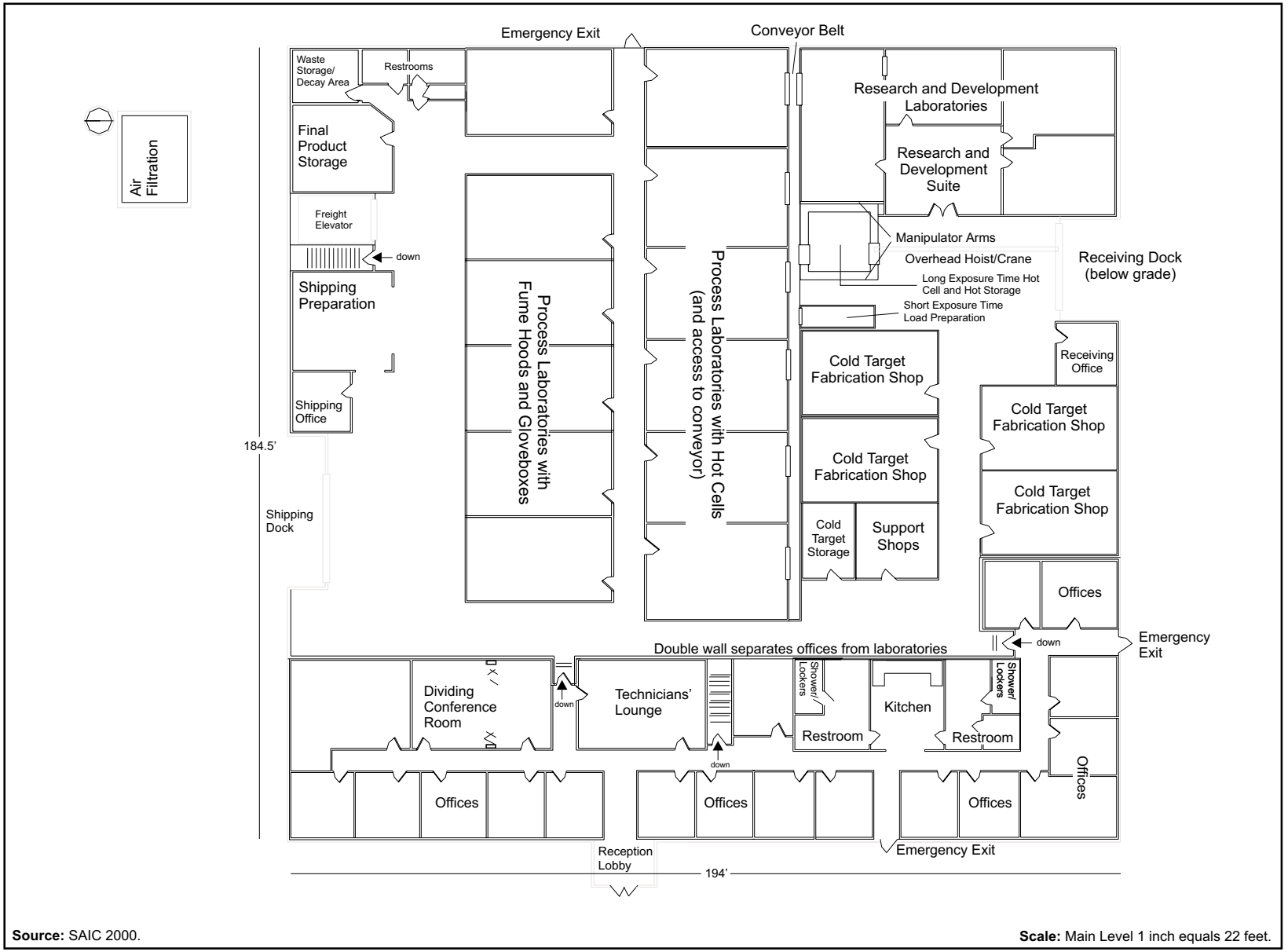
The facility is designed around a center area containing the highest-risk activities and the material inventories requiring the highest level of engineered controls. As can be seen in Figure 2–22, irradiated materials in casks or other shielded transport containers would enter a loading dock with a straight-line access to the primary facility hot cell. The hot sample entry area would be a high bay area with a high floor loading area between the loading dock and the hot cell access port. This configuration would allow transport cask access to the hot cell. In addition, an overhead hoist would be available to facilitate handling of materials and devices in the proximity of the hot cell.

The hot cell would accept high-radiation-level samples or those difficult to shield or manipulate (e.g., reactor core components containing samples). The hot cell would have access to a conveyor that can remotely transport samples to the hot process laboratories. In addition, samples from the hot cell could be transferred to the hot research and development laboratory gloveboxes for detailed analysis and testing. Hot cell manipulators would be located on both the operating gallery and the research and development sides of the hot cell. Adjacent to that would be the central receiving station for all other radioactive and short-exposure samples not contained in the reactor core components. This area, while not a hot cell, would provide personnel protection (i.e., shielding and controlled ventilation) for preliminary sample preparation and examination. It would also provide interim irradiated sample storage prior to delivery to the designated processing laboratory. When needed, samples would be transported remotely to the processing laboratories by the conveyor system. Samples requiring a lesser degree of control would be distributed for processing throughout the remaining process laboratory wing. After processing, the radiopharmaceuticals would be either stored or packaged and shipped immediately to offsite vendors.

Radioactive waste would be packaged and stored for eventual disposal. Those materials containing short-lived isotopes would be delivered to a decay/holding room so that, given appropriate decay time, they could be disposed of without a radioactive component. The process and research and development areas would be considered radiologically controlled areas, but no routinely occupied areas would require control as contaminated radiological areas. Radioactive contamination would be controlled at the hood or glovebox face. Due to this configuration, protective clothing and change rooms would be needed only for occasional maintenance activities when temporary radiological areas are established.

Cold sample (nonradioactive) preparation would be accomplished in a set of three large laboratories where radiological conditions are not anticipated. Completed samples would be stored in an adjacent room along with raw sample materials (nonradioactive).

Radioactive sample preparation and irradiated material recycling activities would be conducted in one of the laboratories adjacent to the conveyor.



Source: SAIC 2000.

Scale: Main Level 1 inch equals 22 feet.

Figure 2–22 Support Facility Layout